

GPU 2205

# Analysis of Three Mile Island - Unit 2 Accident

NSAC-80-1

NSAC-1 Revised March 1980

Prepared by the  
Nuclear Safety Analysis Center

Operated by the  
Electric Power Research Institute  
3412 Hillview Avenue  
P.O. Box 10412  
Palo Alto, California 94303

B307050509 800331  
PDR ADOCK 05000289  
P HOL

NUCLEAR SAFETY ANALYSIS CENTER  
SEQUENCE OF EVENTS  
TMI 2 Accident - March 28, 1979

COMMENTARY

The intent of this sequence of events (SOE) prepared by the Nuclear Safety Analysis Center (NSAC) staff is to present a factual account of the event at the Three Mile Island, Unit 2, nuclear power plant on March 28, 1979. It is not meant to explain why the event occurred or to speculate on why certain actions were taken. This SOE has been verified, when possible, by raw data such as computer output, reactimeter data, strip charts, etc. Events have also been identified and verified by thermal hydraulic or core analysis. Entries concerning radiation readings were the only ones which were not substantiated by raw data because of the unavailability of readable, reproduced or original, strip charts at NSAC. Limited use was made of interviews by others with persons involved in the event. NSAC had no first-hand interviews with TMI-2 operators on duty at the time. Commentary as to why certain actions were taken have been generally omitted, except by inference. Certain actions may have been omitted if they could not be confirmed, even though they have been reported by other sources, as long as these actions did not appear to play an important role in the event.

Appendices have been prepared to further explain the actions and remarks, and to give a narrative description of the events from a thermal hydraulic and core assessment viewpoint. System related appendices are not meant to be independent system descriptions, but are written to clarify or amplify remarks in the SOE.

In some cases, it has been difficult to interpret the available data and the interpretation is still in progress. An example of this is the interpretation of make up pumps status alarms and engineered safeguard actuation status alarms.

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SEQUENCE OF EVENTS  
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Plant Status Prior to Start of Event

TMI Unit 2 was operating at 97% power with the integrated control system (ICS) in full automatic. The reactor coolant system was operating with four reactor coolant pumps at a pressure of 2155 psig. Reactor coolant makeup pump 1B was in service providing normal makeup and reactor coolant pump seal injection flow. Reactor coolant system letdown flow was approximately 70 gpm. The reactor coolant system boron concentration was approximately 1030 parts per million. Rod groups one through five were fully withdrawn, rod groups six and seven were 95% withdrawn and rod group eight was 27% withdrawn. Reactor coolant system leakage was approximately 6 gpm. The condensate system was operating with two condensate pumps and two condensate booster pumps in service. Both turbine driven feed pumps were in service. The pressurizer spray valve and the pressurizer heaters were in manual control while spraying the pressurizer to equalize the boron concentrations between the pressurizer and the rest of the reactor coolant system. This equalization was necessary because of steam leakage past either the electromatic relief valve or the pressurizer safety valves. This leakage was evidenced by periodic safety valve discharge header temperature alarms.

Operators were experiencing difficulties in transferring resins from an isolated condensate polisher to the receiving tank. Attempts to free the plugged transfer line had been in progress for about eleven hours.

TMI Unit 1 was in hot shutdown for low power physics testing following refueling. There was a vacuum on the condenser and auxiliary steam was being supplied from Unit 2.

TIME	EVENT	REMARKS & REFERENCES
-00:00:01 (0400:36)	Condensate pump 1A & 1B (CO-P-1A) tripped.	Ref. 3.a., Appendix C/FDW, p.3.

----- START OF EVENT -----

00:00:00 (0400:37)	Feedwater pumps 1A & 1B (FW-P-1A & FW-P-1B) tripped.
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Ref. 3.a., 1.r, 1.s. The trips were caused by low feedwater pump suction pressure. Feedwater flow was lost to both steam generators.

00:00:00 (0400:37)	The main turbine tripped.
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Ref. 3.a. The turbine trip results automatically from the trip of both feedwater pumps.

00:00:00 (0400:37)	Three emergency feedwater pumps 1, 2A & 2B (EF-P-1, EF-P-2A, and EF-P-2B) started.
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Ref. 3.a. Automatic start of these pumps is caused by the trip of both feedwater pumps. Emergency feedwater pump 1 (EF-P-1) is a steam turbine driven pump, and emergency feedwater pumps 2A & 2B (EF-P-2A and EF-P-2B) are electric motor driven pumps.

00:00:03 (0400:40)	The pressure setpoint (2255 psig) of electro-matic relief valve (ERV) (RC-2) located on the pressurizer was exceeded.
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Ref. 1.k, 5.b. The rate-of-change of the reactor coolant system pressure indicated that the relief valve lifted.



TIME	EVENT	REMARKS & REFERENCES
00:00:04 (0400:41)	Pressure began increasing in the reactor coolant drain tank.	Ref. 1.j. This was another indication that the ERV lifted. Reactor coolant drain tank pressure at this time was approximately 4 psig.
00:00:06 (0400:43)	Secondary side steam pressure reached 1074 psig in steam generator A and 1052 psig in steam generator B.	Ref. 1.u, 1.v. Pressures from the reactimeter charts indicated that some main steam safety valves lifted.
00:00:08 (0400:45)	The reactor tripped on high reactor coolant system pressure. The nominal trip setpoint is 2355 psig.	Ref. 3.a, 1.k, 11 - Sect. 2.2.1, Appendix TH, p. 27. Reactor coolant system pressure reached 2344 psig at 0400:46, as indicated on the reactimeter. The higher reading of approximately 2435 psig observed on the wide range reactor coolant system stripchart could be caused by a 50 to 60 psig higher reading at the start of the event coupled with pen overshoot during the rapid pressure increase.
00:00:08 (0400:45)	Pressurizer heater groups 1 through 5 indicated off.	Ref. 3.a, 2.e - p. 2, Appendix RCPCS - pp. 2-5. This event could signify the operator's reported action of placing the pressurizer heater control in the automatic mode to mitigate expected reactor coolant system pressure transients following any reactor trip.

TIME	EVENT	REMARKS & REFERENCES
00:00:12 (0400:49)	Reactor coolant system pressure decreased below the setpoint value for ERV (RC-R2) closure.	Ref. 1.k, 5.b, Appendixes ERV - p. 1, RCPCS - pp. 3, 6-7. The ERV should have reseated (closure setpoint was 2205 psig), but it remained in the full open position.
00:00:12 (0400:49)	Indicated pressurizer coolant level peaked at 256 in. and began a rapid decrease.	Ref. 1.g. A momentary coolant insurge followed by a rapid coolant outsurge is a normal event following a reactor trip.
00:00:12 (0400:49) Approximate	Letdown flow was stopped.	Ref. 3.d, 2.d, - p. 1, 2.e - p. 2, Appendix HPI - pp. 1, 3, 6. This step and the next are initial actions for operators to take after reactor trip accidents to compensate for the expected reduction of pressurizer level.
00:00:13 (0400:50)	One or more attempts were made to start makeup pump 1A (MU-P-1A). The pump did not start.	Ref. 3.a., 3.d, 2.c - p. 2, 2.d. - p. 2, Appendix HPI - pp. 1, 6, 7.
00:00:13 (0400:50)	A condenser hotwell low water level alarm was received on the alarm typewriter. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.a, Appendixes C/FDW - p. 5, FDS - pp. 12-13. The level was 21.72 in. Because of the 15 second scan interval of this parameter, the actual alarm may have been received before this time.

TIME	EVENT	REMARKS & REFERENCES
00:00:14 (0400:51)	The emergency feedwater pumps (EF-P1, EF-P-2A, and EF-P-2B) low pressure alarms cleared on the alarm typewriter.	Ref. 3.a, Appendix PDS - pp. 12-13. EF-P1            1237 psig EF-P2-A        1471 psig EF-P2-B        1445 psig  Because of the 15 second scan interval of these parameters, these pressures were probably reached before the time printed out.
00:00:14 (0400:51)	Pressurizer heater groups 1 through 5 indicated on.	Ref. 3.a, 2.e - p. 2, Appendix RCPCS - pp. 2, 3-5. A low reactor coolant system pressure with the heater control in automatic would cause the heaters to energize.
00:00:15 (0400:52)	Steam generator A water level indicated 74 in. on the startup range. Steam generator B water level indicated 76 in. on the startup range.	Ref. 1.q, 1.x, Appendix OTSG - pp. 2-3. Steam generator water levels were decreasing.
00:00:15 (0400:52)	The pressurizer spray valve (RC-V1) closed).	Ref. 1.i, 2.e - p. 2, Appendix RCPCS - pp. 5-6. This closure indicated that the spray valve was responding normally to reactor coolant system pressure transients and supports the operator's reported action of placing the pressurizer spray in automatic.

TIME	EVENT	REMARKS & REFERENCES
00:00:28 (0401:07)	The condenser hotwell low water level alarm cleared. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.a, Appendix C/FDW - p. 5. The level was 26.44 in. The computer scan interval was 15 sec.
00:00:30 (0401:07)	The ERV (RC-R2) and pressurizer safety valve (RC-R1B) outlet temperatures alarmed high.	Ref. 3.c, Appendix ERV - pp. 3-4. RC-R2            292.2°F RC-R1B          203.5°F RC-R1B outlet temperature had been operating close to its high alarm setpoint prior to the reactor trip. Computer scan interval was 30 sec.
00:00:30 (0401:07)	The reactor coolant system low pressure trip setpoint was reached.	Ref. 1.q, 5.b, Appendices OTSG - pp. 2-4, PDS, p.2. Emergency feedwater valves EF-V11A & EF-V11B should open when levels reach 30 inches. Feedwater was not admitted to the steam generators. Emergency feedwater block valves EF-V12A and EF-V12B, which should have been open, were closed. This fact was not recognized by the operators at this time.
00:00:33 (0401:10)	Steam generator B water level decreased to 28.8 in. in the startup range.	Ref. 1.x, 5.b, Appendices OTSG - pp. 2-3, PDS - p.2. See entry at 0401:07.
00:00:41 (0401:18)	Makeup pump 1A (MU-P-1A) was started.	Ref. 3.a, 1.q. With makeup pumps 1A & 1B operating, the pressurizer level rate of decrease slowed.

TIME	EVENT	REMARKS & REFERENCES
00:00:48 (0401:25)	Pressurizer coolant reached an indicated minimum level of 158 in.	Ref. 1.g. Indicated pressurizer coolant level began to increase.
00:00:58 (0401:35)	A pressurizer low coolant level alarm was received. (Nominal alarm setpoint was 200 in.)	Ref. 3.a, Appendix PDS - pp. 2, 15, 12-13. Because of the scan intervals and computation involved with pressurizer coolant level, this alarm typewriter entry lagged the actual pressurizer low level.
00:01:00 (0401:37)	Pressurizer safety valve (RC-R1A) outlet temperature alarmed high.	Ref. 3.a, Appendix RCPCS - pp. 3, 5-6. RC-R1A 294.5°F
00:01:13 (0401:50)	A condenser hotwell high water level alarm was received. (Nominal alarm setpoint was 36 in.)	Ref. 3.a, 2.C - p. 3, 2.a - p. 2, Appendix C/FW - p. 5. Indicated level was 37.77 in. The computer scan interval was 15 sec.
00:01:26 (0402:03)	The indicated reactor coolant drain tank liquid temperature was 85.5°F.	Ref. 3.a. This alarm typewriter entry indicated that the reactor coolant drain tank liquid returned to a normal temperature range. Because of the heat addition to the tank at this time, it is probable that it was a low temperature alarm that had cleared.

TIME	EVENT	REMARKS & REFERENCES
00:01:45 (0402:22) Approximate	Both steam generators boiled dry on the secondary side.	Rrf. 1.b, 1.c, 1.d, 1.e, 1.u, 1.v. This event was indicated by a steadily decreasing steam generator secondary side pressure while reactor coolant hot and cold leg temperatures were increasing. Effective heat transfer from the primary coolant to the secondary system is stopped when the steam generators are dry.
00:02:01 (0402:38)	Engineered safety features for high pressure injection actuated.	Ref. 3.a, 2.c - p. 2, 2.e - p. 4, II Sec. 3.3-2-1, Appendices ESF - pp. 6, 7, 9, HPI - p. 9. This was caused by reactor coolant system pressure dropping below 1640 psig.
	Makeup pump 1B (MU-P-1B) tripped.	Ref. 3.a, 2.g - p. 2, Appendix ESF - pp. 14-15. This is an automatic trip before high pressure injection pumps 1A & 1C (MU-P-1A and MU-P-1C) start.
through	High pressure injection pump 1C (MU-P-1C) started automatically.	Ref. 3.a, 2.g - p. 2, Appendices ESF - pp. 9, 14, HPI - p. 9. Pump 1A (MU-P-1A) was already operating.

TIME	EVENT	REMARKS & REFERENCES
00:02:04 (0402:41)	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) started.	Ref. 3.a, Appendix ESP - pp- 2, 16. These pumps were running in a recirculation mode and were not injecting water into the reactor coolant system.
00:03:13 (0403:50)	The high pressure injection portion of engineered safety features was manually bypassed.	Ref. 3.a, 2.c - p. 2, 2.g - p. 2, Appendices ESP - p. 12, HPI - p. 9. This action permits throttling discharge valves of the high pressure injection pumps and/or shutting down the high pressure injection pumps.
00:03:13 (0403:56)	The reactor coolant drain tank relief valve (MOL-R1) lifted at approximately 122 psig. Nominal setpoint is 150 psig.	Ref. 1.j. This is an inferred event based on the reactor coolant drain tank pressure history. A lifted pressurizer relief or safety valve could cause the reactor coolant drain tank relief valve to lift.
00:03:25 (0404:03)	A reactor coolant drain tank high temperature alarm was received.	Ref. 3.a. This is another indication of a lifted pressurizer relief or safety valve. Indicated temperature was 127°F. Computer scan interval was 30 sec.
00:03:29 (0404:05)	A pressurizer high coolant level alarm was received. (Nominal alarm setpoint was 260 in.)	Ref. 3.a, 1.g, Appendices HPI - p. 4, PDS - pp- 2, 4, 12-13. The pressurizer coolant level as indicated by the reactimeter was 292.4 in. Computer scan interval was 15 sec.

TIME	EVENT	REMARKS & REFERENCES
00:04:38 (0405:15)	The operator stopped makeup pump 1C (MU-P-1C) and throttled the high pressure injection isolation valves (MU-V16A & MU-V16B).	Ref. 3-a, Appendix HPI - pp. 4, 9, 2.d - p. 5, 2-m - p. 1. The pressurizer coolant level rate of rise decreased. Hot leg temperatures started to increase, followed by an increase in cold leg temperatures.
00:04:52 (0405:29)	Intermediate closed cooling pump 1A (IC-P-1A) started.	Ref. 3-a. This is a normal procedural step prior to initiating high letdown flow.
00:04:58 (0405:35)	An alarm was received that letdown flow was off scale.	Ref. 3-a, 1.h, 2.c - p. 3, 2-m - p. 2. This alarm could indicate that letdown flow was off scale high, greater than 160 gpm.
00:05:00 (0405:37)	Pressurizer coolant level reached 377 in., decreased momentarily, then continued to rise.	Ref. 1-g.
00:05:15 (0405:52)	Condensate pump 1A (CO-P-1A) was started.	Ref. 3-a, Appendix C/FDM - p. 3.
00:05:15 (0405:52)	A condensate booster pump 2B (CO-P-2B) trip signal was received three times, followed by a trip clearing (normal) signal in each through	Ref. 3-a, Appendix C/FDM - p. 4. Operators were trying to reestablish secondary plant operating conditions. The inability to start the booster pump was apparently caused by a low suction pressure trip when the pump started turning.
00:06:29 (0407:06)	instance.	



TIME	EVENT	REMARKS & REFERENCES
00:05:30 (0406:07) Approximate	The indicated reactor coolant system hotleg temperature and pressure reached saturation conditions of 582°F and 1340 psig.	Ref. 3.d, 4.f, 1.b, 1.c.
00:05:46 (0406:23)	There were indications of a liquid discharge from the ERV.	Ref. 1.j. This was evidenced by a rapid increase in the reactor coolant drain tank pressure which indicated increased mass flow corresponding to liquid flow from the ERV.
00:05:51 (0406:28)	Pressurizer coolant level indication went off scale high, greater than 400 in.	Ref. 1.g.
00:06:54 (0407:31)	A letdown cooler 1A high temperature alarm was received. (Nominal alarm setpoint was 135°F.)	Ref. 3.a, 3.d, 1.h, Appendix HPI - p. 12. This alarm could be associated with high letdown flow. Indicated temperature was 139°F. The computer scan interval was 30 sec. Letdown flow is automatically isolated when this alarm is received by closure of the letdown isolation valve (MU-V-376).
00:06:58 (0406:35)	Letdown flow came back on scale and indicated 71.4 gpm.	Ref. 3.a, Appendix HPI - p. 12. Letdown flow was decaying following closure of the letdown isolation valve. Computer scan interval was 30 sec.
00:07:29 (0409:06)	Reactor building sump pump 2A (WDL-P-2A) started.	Ref. 3.a, 2.c - p. 4, Appendices SP, ROUTES - p. 2.

TIME	EVENT	REMARKS & REFERENCES
00:08:13 (0408:55)	Emergency feedwater block valves 12A & 12B (EF-V12A and EF-V12B) were opened.	Ref. 1.b, 1.c, 1.d, 1.e, 1.g, 1.u, 1.v & 1.x, 2.c - p. 2, 2.d - p. 3, 2.e - p. 3, 2.a - p. 1. Normally these valves should have been open. Opening them admitted water to the steam generators. Increases in steam generator levels and pressures followed by responses of hot and cold leg temperatures indicate feed was established to the steam generators.
00:08:30 (0409:07) Approximate	The reactor coolant system pressure began to decrease further.	Ref. 4.d, Appendix G/SG - p. 3. The initiation of cold emergency feedwater to the steam generator resulted in rapid decreases of the reactor coolant system temperatures and pressures.
00:08:58 (0409:35)	Condensate pump 1A (CO-P-1A) tripped.	Ref. 3.a, Appendix C/TDW - p. 3.
00:09:13 (0409:50)	A condensate booster pump suction header low pressure alarm was received.	Ref. 3.a, Appendix C/TDW - p. 4. The pressure was 14.7 psig.
00:09:23 (0410:00) Approximate	Letdown flow isolation valve (MO-V-376) was opened.	Ref. 3.d, 1.b.

TIME	EVENT	REMARKS & REFERENCES
00:10:15 (0410:52)	Pressurizer coolant level indication came back on scale, less than 400 in.	Ref. 1.g. The initiation of emergency feedwater to the steam generators could result in coolant out-surges from the pressurizer.
00:10:19 (0410:56)	Reactor building sump pump 2B (WDL-P-2B) started.	Ref. 3... Appendix SP.
00:10:24 (0410:56)	Letdown cooler outlet temperature returned to its normal range and the alarm cleared. (Nominal alarm setpoint was 135°F.)	Ref. 3.a. The temperature was 123.7°F indicating that letdown flow had been reestablished. The computer scan interval was 30 sec.
00:10:24 (0411:01) through 00:11:43 (0412:29)	Makeup pump 1A (MU-P-1A) tripped and was re-started three times.	Ref. 3.a. The pump was operating at the end of this sequence.
00:10:49 (0411:25)	A reactor building sump high level alarm was received.	Ref. 3.a, Appendix SP. Indicated level was 4.65 feet.
00:13:13 (0413:50)	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) were shut down.	Ref. 3.a, Appendix ESP - p. 2. These pumps served no function at that time.

TIME	EVENT	REMARKS & REFERENCES
00:13:27 (0414:04)	The condensate booster pump section header low pressure alarm cleared. (Nominal alarm setpoint was 15 psig.)	Ref. 3-a. The pressure was 17 psig. The computer scan interval was 15 sec.
00:14:48 (0415:25)	The reactor coolant drain tank rupture disc (WDL-026) failed at 191.5 psig.	Ref. 1-j, Appendix B00TES - p. 1. Design failure pressure is 200±25 psig. Discharge through the ruptured disc was to the reactor building atmosphere.
00:14:50 (0415:27)	At this time, reactor coolant pump related alarms began coming in on the alarm typewriter.	Ref. 3-a. These alarms could indicate abnormal reactor pump operation caused by degraded conditions in the reactor coolant system and reactor building. Reactor coolant flow had been steadily decreasing since the beginning of the event.
00:15:43 (0415:20)	The condensate booster pumps low discharge pressure alarm was received.	Ref. 3-a, Appendix C/FDW - p. 5. Discharge pressure was 307 psig.
00:16:12 (0416:49)	A condensate booster pump section header low pressure alarm was received. (Nominal alarm setpoint was 310 psig.)	Ref. 3-a, Appendix C/FDW - p. 4. Indicated pressure was 14.8 psig. The computer scan interval was 15 sec.

TIME	EVENT	REMARKS & REFERENCES
00:19:23 (00420:00) Approximate	Reactor Building Purge Air Exhaust Duct A Monitor (HP-R-225) particulate channel count rate increased from $1 \times 10^2$ to $5 \times 10^2$ cpm. Slight increases were also indicated on the Duct B Monitor (HP-R-226) and on HP-R-222 (before the filter) and HP-R-228 (after the filter).	Ref. 12.a, Appendix HM - p. 3. These indications appear to be the result of the reactor coolant drain tank rupture disc blowout.
00:20:00 (00420:37) Approximate	The indicated source range neutron flux signal departed from the expected normal flux decay for a reactor trip.	Ref. 4.f. This departure was caused by the buildup of steam voids in the system. Source range count rate leveled and then began a gradual increase.
00:22:17 (00422:54)	The operator depressed the reactor trip push-button.	Ref. 3.a, Appendix CI - p. 12. This precautionary action was in response to the observed abnormalities in neutron flux levels.
00:22:44 (00423:21)	Steam generator A water level increased to approximately 30 in. in the startup range. The low level alarm cleared.	Ref. 3.a, 1.q, Appendix OTSC - pp. 2-3. The design condition for heat removal from the steam generator was reestablished.
00:24:50 (00425:35)	The RRV outlet temperature was 285.4°F. Safety valves R1A and R1B read 263.9° and 275.1° respectively.	Ref. 3.c, Appendix BCPCS - pp. 3, 6-7. This was operator requested information.

TIME	EVENT	REMARKS & REFERENCES
00:25:44 (025:21)	An emergency feedwater pump 1 (EF-P-1) low discharge pressure alarm was received.	Ref. 3-a. Discharge pressure was 9 psig. This could indicate that emergency feed pump 1 was shut down. No other status indication was available for this pump.
00:26:26 (0427:03)	EC Loop A outlet temperature 551.9°F	Ref. 3-c, Appendix PDS - pp. 1-3, 13-14. This was operator requested plant status information on the utility typewriter. Reactor temperatures are in close agreement with these values.
through	EC Loop B outlet temperature 550.9°F	
00:27:51 (0427:28)	EC Loop A inlet temperature 548.1°F	
	EC Loop A inlet temperature 547.0°F	
	EC Loop B inlet temperature 547.0°F	
	EC Loop B inlet temperature 546.8°F	
	EC Loop A wide range pressure 1040 psig.	
	EC Loop B wide range pressure 1043 psig.	
00:26:06 (0427:33)	Steam generator B water level increased to approximately 28 in. in the startup range. The low level alarm cleared.	Ref. 3-a, 1-x, Appendix OTSG - pp. 2-3. Conditions for effective heat removal from the steam generator were reestablished.

TIME	EVENT	REMARKS & REFERENCES
00:29:23 (0430:00) through 00:49:23 (0450:00) Approximate	Reactor Building Air Sample Monitor (RP-R-227) gas channel count rate increased from $1 \times 10^3$ to $5 \times 10^4$ cpm and then decreased to $1 \times 10^3$ cpm.	Ref. 12-a, Appendix III - p. 3.
00:29:56 (0430:33)	Both emergency diesel generators were manually tripped during the next 30 sec.	Ref. 3-a.
00:32:23 (0433:00) Approximate	Radiation readings of the following monitors increased and then leveled off: gas channel of Station Vent (RP-R-221A, -221B) H <sub>2</sub> Purge Duct particle and iodine channels (RP-R-229).	Ref. 12-a, Appendix III - p. 3.
00:32:36 (0433:13)	Incore thermocouple (Location 10-R) indicated off scale.	Ref. 3-a, Appendix CII - p. 16. Top of the scale is 700°F.
00:36:00 (0436:45)	Emergency feedwater pump 2B (RP-P-2B) was shut down.	Ref. 3-a, 1-q, 1-x. Steam generator A water level indicated 35.3 in. in the startup range. Steam generator B water level indicated 40.3 in. in the startup range. These levels are normal for shutdown conditions. Emergency feedwater pump 2A was running to maintain shutdown levels in the steam generators.

TIME	EVENT	REMARKS & REFERENCES
00:38:19 (0438:47)	Reactor building sump pump 2A (WDL-P-2A) was stopped.	Ref. 3.a, 2.c - p. 4, 2.f - p. 4, Appendix SP. This pump ran for approximately 31 min.
00:38:11 (0438:43)	Reactor building sump pump 2B (WDL-P-2B) was stopped.	Ref. 3.a, 2.c - p. 4, 2.f - p. 4. All pumping of water from the reactor building to the auxiliary building was stopped. This pump ran for approximately 28 min.
00:40:00 (04:40:37) Approximate	An increasing count rate continued to be indicated on the source range neutron detector.	Ref. Appendix CI - p. 12. This increase was caused by the decreasing density of the coolant passing through the reactor downcomer annulus which shields the core.
00:46:23 (0447:00) Approximate	Letdown Cooler A Monitor (IC-R-1092) count rate began increasing from approximately 2000 cpm and reached over $2 \times 10^4$ cpm about 40 minutes later.	Ref. 12.a, 2.c - p. 4, 2.e - p. 7, Appendices RM - p. 1, ROUTES - p. 3.
00:59:12 (0459:49)	The condensate booster pump suction header low pressure alarm cleared. (Nominal alarm set-point was 15 psig.)	Ref. 3.a, Appendix C/PDW - p. 4. Pressure was 89.2 psig. The computer scan interval was 15 sec.
00:59:21 (0459:58)	A condensate high temperature alarm was received.	Ref. 3.a, Appendix C/PDW - p. 4-5. Indicated temperature was 118.5°F.



TIME	EVENT	REMARKS & REFERENCES
01:00:49 (0501:26)	Condenser circulating water pumps 1B, 1C, 1D, & 1E (CW-P-1B, CW-P-1C, CW-P-1D, & CW-P-1E) were shut down.	Ref. 3.a, 2.d - p. 9, 2.f - p. 2, 2.n - p. 4, Appendix STEAM DUMP - pp. 2-3, 5. Steam flow control was shifted from the turbine bypass valves to the atmospheric dump valves.
01:13:23 (0514:00)	The alarm history was lost until 0648:08.	Ref. Appendix PDS - pp. 12-13.
01:13:29 (0514:06)	Reactor coolant pump 2B (RC-P-2B) was stopped.	Ref. 3.f, 1.m, 2.c - p. 5, 2.d - p. 5, 2.e - p. 5, 2.n - p. 6. Secondary side steam pressure in loop B began to drop sharply, indicating stagnation in reactor coolant system loop B flow.
01:20:31 (0521:08)	Pressurizer Relief and Safety Valve T	Ref. 3.f, Appendix ERV - pp. 3-5, RCPCS - pp. 3, 6-7. This was an operator requested computer printout of relief and safety valve outlet temperatures. It was not recognized that the ERV temperature indicated that the relief valve was open.
through	Temperatures	
01:20:58 (0521:35)	ERV RC RV2 - 283°F Safety valve RC R1A - 211°F Safety valve RC R1B - 218°F	
01:30:00 (0530:37) Approximate	Reactor out-of-core intermediate range neutron instrumentation channel MI-3 came on scale and began increasing.	Ref. 4.f, 2.d - p. 6, Appendix CI - p. 12. This response was consistent with the steadily increasing source range count rate.

TIME	EVENT	REMARKS & REFERENCES
01:30:40 (0531:17)	There was a marked increase in secondary side steam flow from steam generator A.	Ref. 1.q, 1.a. The combination of changes in steam generator secondary side pressure and water level indicates steam flow.
01:31:22 (0531:59)	Secondary side steam flow from steam generator A decreased rapidly.	Ref. 1.u. The combination of changes in steam generator secondary side pressure and water level indicates steam flow.
01:32:04 (0532:41)	Feedwater flow to steam generator B was increased.	Ref. 1.x. This served no apparent purpose because flow in the reactor coolant system loop B had been stopped.
01:32:19 (0532:56)	Steam generator A indicates dryout on the secondary side.	Ref. 1.q, 1.u. The conditions for effective heat removal using the steam generators had been lost because there was no reactor coolant system flow in loop B and steam generator A was dry.
01:34:10 (0534:47)	Feedwater flow to steam generator A was increased.	Ref. 1.u, 2.c - p. 6, 2.d - p. 5. This was an apparent effort to regain conditions for heat transfer from the reactor coolant system using the steam generators.
01:34:16 (0534:53)	Loop A cold leg temperatures started to decrease.	Ref. 1.d. This was an indication of reestablishment of heat transfer from the reactor coolant system.

TIME	EVENT	REMARKS & REFERENCES
01:34:16 (0534:53)	Feedwater flow to steam generator B was reduced.	Ref. 1.x, 2.c - p. 5.
01:40:37 (0541:14)	Reactor coolant pump 2A (RC-P-2A) was stopped.	Ref. 3.f, 1.f., 2.c - p. 5, 2.e - p. 5. Appendix PDS - pp. 1-2. The reactimeter showed a marked decrease in reactor coolant flow beginning at 0540:57.
01:40:45 (0541:22)	Reactor coolant pump 1A (RC-P-1A) was stopped.	Ref. 3.f, 1.f, 2.c - p. 5, 2.3 - p. 5, Appendix PDS - pp. 1, 2. At this point there was no forced reactor coolant system flow. The reactimeter showed a marked flow decrease beginning at 0541:08.
01:41:00 (0541:37) Approximate	Out-of-core neutron instrumentation indicated a decreasing flux level.	Ref. 4.f, Appendix CI - p. 13. This temporary decrease was caused by a coolant phase separation with the liquid filling the downcomer annulus. The phase separation is attributed to stopping the reactor coolant pumps and the resultant flow coastdown.
01:42:00 (0542:37)	Steam generator B was isolated on the secondary side.	Ref. 1.v, 1.x, 2.c - p. 5, 2.m - p. 7, Appendices OTSG - p. 4, TH - p. 52. A leak was suspected because of difficulties with water level control in steam generator B and a 300 psi pressure differential between steam generators A and B.

TIME	EVENT	REMARKS & REFERENCES
01:42:25 (0543:02)	Reactor coolant system loop A cold leg temperature stopped decreasing and small temperature oscillations began.	Ref. 1.d. This could indicate stopping circulation of coolant through the loops.
01:42:30 (0543:07) Approximate	Out-of-core nuclear instrumentation indicated increased flux levels.	Ref. 4.f, Appendix CI - p. 13. These indicated higher flux levels could be caused by the boil-off of the coolant in the core and the resultant decrease in coolant level in the downcomer annulus.
01:51:27 (0552:04) through 02:29:18 (0629:55)	Loop A and B hot leg temperatures were increasing and continued upward until they went off scale high, greater than 620°F.	Ref. 1.b, 1.c, 2.c - p. 6, Appendix TH - p. 53. These temperatures and the reactor coolant system pressure indicate the presence of superheated steam in the reactor coolant system.
02:03:57 (0604:34)	An operating range water level of 51% was established and subsequently maintained in steam generator A.	Ref. 1.p, 1.q, 2.c - p. 6. This is a requirement for establishing natural circulation in the reactor coolant system.

TIME	EVENT	REMARKS & REFERENCES
02:14:23 (0615:00) Approximate	The Reactor Building Air Sample Monitor (HP-R-227) particulate channel radiation reading increased and eventually went off scale high. The gas channel radiation reading began to increase at 0625 and then went off scale high. Iodine channel count rate began to increase about 0645 and went off scale high.	Ref. 12.a, Appendix EM - p. 3.
02:15:00 (0615:37)	Self powered neutron detector readings began to rise rapidly.	Ref. 4.h, Appendix CI - p. 21. Because the core was partially uncovered, the self powered neutron detectors were responding as local temperature detectors which indicated high core temperatures. This response was because the self powered neutron detectors are also thermionic emitters that generate a measurable signal at high temperature.
02:17:53 (0618:30)	ERV (RC-R2) outlet temperature was 228.7°F.	Ref. 3.e, 2.r - p. 1, Appendix ERV - pp. 3-5. The ERV outlet temperature was operator requested.

TIME	EVENT	REMARKS & REFERENCES
02:22:00 (0622:37) Approximate	The ERV block valve (RC-V2) was closed. Reactor coolant system pressure began to increase.	Ref. 4.a, 4.d., 2.c - p. 5, 2.m - p. 6, 2.4 - p. 2. Reactor coolant leakage through the ERV was stopped. Decreasing reactor building pressure and increasing reactor coolant system pressure indicated the primary source of system leakage was through the ERV.
02:33:27 (0634:04)	Increasing the level in steam generator B was begun.	Ref. 1.w, 1.x, Appendix OTSG - pp. 2-3. Steam generator B level was 5% on the operating range.
02:34:23 (0635:00) Approximate	An additional makeup pump was started.	Ref. Appendix TH - p. 60.
02:38:23 (0639:00) Approximate	The Makeup Tank Area Monitor (HP-R-206) (in the auxiliary building), Fuel Handling Bridge South Monitor (HP-R-210) in the reactor building, and Reactor Building Dome Monitor (HP-R-214) radiation readings began to increase.	Ref. 12.a, Appendix NM - p. 2.
02:38:23 (0639:00) Approximate	Letdown Cooler A Radioactivity Monitor (IC-R-1092) readings pegged off scale high.	Ref. 12.a, Appendix NM - p. 1.

TIME	EVENT	REMARKS & REFERENCES
02:39:23 (0640:00) Approximate	Two boron analyses indicating boron concentrations in the reactor coolant system of approximately 400 ppm were received by the shift supervisor. Emergency boration of the reactor coolant system was started.	Ref. 2.a - p. 3, 2.b - p. 2, 2.m - p. 5, 4.f. These analyses, in conjunction with the increased neutron level indications, prompted emergency boration to maintain the reactor subcritical.
02:44:23 (0645:00) Approximate	The In-core Instrument Panel Area Radio-activity monitor (HP-R-213) (in the reactor building) radiation readings increased off scale high.	Ref. 12.a, Appendix EM - p. 2.
02:44:23 (0645:00) Approximate	Reactor Building Purge Air Exhaust Ducts A & B Monitors (HP-4-225 & HP-4-226) particulate and gas channels radiation readings increased rapidly.	Ref. 12.a, Appendix EM - p. 3.
02:44:23 (0645:00) Approximate	The makeup pump started at 0635 was stopped.	Ref. Appendix TH - p. 61.
02:45:23 (0646:00) Approximate	Fuel Handling Storage Area Monitor (HP-R-218) radiation readings began increasing.	Ref. 12.a, Appendix EM - p. 2.

TIME	EVENT	REMARKS & REFERENCES
02:46:23 (0647:00)	An attempt was made to start reactor coolant pump 1A (RC-P-1A).	Ref. 2.a - pp. 3 & 13, 2.c - p. 6, 2.d - p. 6. The pump did not start.
02:47:31 (0648:08)	Current alarm typewriter indications showed that self-powered neutron detectors were responding to high temperatures down to the four foot level of the core (approximate). Ninety percent of the core exit thermocouples were reading in excess of 700°F.	Ref. 3.g, Appendix CI - p. 16. These were the first alarms that were received following the period when alarm history was lost.
02:49:23 (0650:00) Approximate	The radiation readings of all channels of the following monitors increased steadily and by 0721 were off-scale: Station Vent (HP-R-219), Fuel Handling Building Exhaust Duct (HP-R-221 A & B), Hydrogen Purge (HP-R-229). The radiation readings of all channels of Control Room Intake Monitor (HP-R-220) remained below 10 cpm from 0000 hours March 28 until 0950 hours March 28.	Ref. 12.a, Appendix FM - p. 3.



TIME	EVENT	REMARKS & REFERENCES
02:49:23 (0650:00)	Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) radiation readings increased rapidly from $1 \times 10^2$ to $8 \times 10^5$ cpm.	Ref. 12.a, Appendix RM - p. 3. This monitor, located in the turbine building on the 281 ft - 6 in level, samples condenser exhaust from the vacuum pumps, and the readings could be indicative of a primary to secondary leak.
02:59:23 (0700:00)	At approximately 0850 this monitor's radiation reading decreased to $1 \times 10^4$ cpm.	
Approximate		
02:51:57 (0652:34)	The operator attempted to start reactor coolant pump 2A (RC-P-2A).	Ref. 3.g, 2.d - p. 6, 2.a - pp. 3 & 13, 2.c - p. 6. The pump would not start. Indications show that preliminary steps were taken to start reactor coolant pump 2A (RC-P-2A).
02:52:30 (0653:07)	The condenser hotwell high water level alarm cleared. (Nominal alarm setpoint was 36 in.)	Ref. 3.g. Indicated level was 34.94 in. The computer scan interval was 15 sec.
02:53:16 (0653:53)	The operator attempted to start reactor coolant pump 1B (RC-P-1B).	Ref. 3.g, 2.d - p. 6, 2.a - pp. 3 & 13, 2.c - p. 6. The pump would not start. Indications show that preliminary steps were taken to start reactor coolant pump 1B (RC-P-1B).
02:54:09 (0654:46)	The operator started reactor coolant pump 2B (RC-P-2B). Flow was indicated for only a few seconds and then returned to 0.	Ref. 3.g, 1.d, 1.e, 2.d - p. 6, 2.a - pp. 4 & 13, 2.c - p. 6, 2.e - p. 6, Appendix TH - pp. 61 - 63. Forced reactor coolant system flow was reestablished. Reactor coolant pump 2B (RC-P-2B) was running with high vibration alarm.

TIME	EVENT	REMARKS & REFERENCES
02:54:15 (0654:52) Approximate	The reactor out-of-core nuclear instrumentation showed sharp neutron flux decreases, followed by increases which approach the levels prior to the starting of reactor coolant pump 2B (RC-P-2B).	Ref. 4.f, Appendix CI - p. 12. The downcomer annulus was temporarily filled with coolant from the cold leg piping, shielding the detector from the core.
02:54:19 (0654:56)	Pressurizer heater groups 1 through 5 tripped.	Ref. 3.g, 2.g - p. 7, Appendix RCPCS - pp. 2, 3 - 5.
02:54:23 (0655:00) Approximate	Waste Gas Discharge Monitor radiation readings (WDG-R-1480) began to increase and went off scale high.	Ref. 12.a, Appendices RM - p. 3, AUX BLDG. This is at the 305 ft elevation in the auxiliary building.
02:54:50 (0655:27)	High pressure injection engineered safety features actuation logic automatically reset on increasing reactor coolant system pressure.	Ref. 3.g, 1.k, Appendix ESP - p. 12.
02:54:50 (0655:27)	Circulating water pump 1B (CW-P-1B) was started.	Ref. 3.g.
02:55:00 (0655:37) Approximate	A site emergency was declared.	Ref. 2.c - p. 7, 2.d - pp. 6 & 7, 2.a - p. 4, 2.b - p. 6, 2.m - p. 2, 2.p - p. 3, 2.n - pp. 2 & 3. Notification of offsite authorities was begun.

TIME	EVENT	REMARKS & REFERENCES
02:55:13 (0655:50)	The engineered safety features bypasses were cleared.	Ref. 3.g, Appendix ESP - p. 12.
02:55:26 (0656:03)	A condenser hotwell low water level alarm was received. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.g. Indicated level was 21.82 in. The computer scan interval was 15 sec.
02:55:38 (0656:15)	Circulating water pump 1E (CW-P-1E) was started.	Ref. 3.g.
02:56:12 (0656:49)	Main steam isolation valves MS-V4B & MS-V7B indicated open.	Ref. 3.g.
02:56:19 (0656:56)	Main steam isolation valves MS-V4B and MS-V7B indicated closed.	Ref. 3.g, 2.c - p. 7. Stroke time for these valves is approximately 117 seconds.
02:59:23 (0700:00) Approximate	The Fuel Handling Building Air supply fan stopped and remained off the remainder of March 28.	Ref. 12.a.

TIME	EVENT	REMARKS & REFERENCES
(0700:00 March 28 to 1100:00 April 2) Approximate	Radiation monitors HP-R-222, 225, 226, 228 were off scale high or nearly off scale during this period.	Ref. 12-a, Appendix FM - p. 3.
03:00:00 (0700:37)	Reactor coolant system pressure was approximately 2945 psig.	Ref. 1.k.
03:00:56 (0701:33)	Condensate hotwell water level was off scale low.	Ref. 3-g, Appendix C/FDW - p. 5. The scale indicates from 10 to 50 in.
03:01:11 (0701:49)	The condensate storage tank B low water level alarm cleared. (Nominal alarm setpoint was 20 ft.)	Ref. 3-g, Appendix C/FDW - p. 6. Indicated level was 20.31 ft. The computer scan interval was 15 sec.
03:02:56 (0703:33)	A condenser hotwell low water level alarm was received, indicating that level had come back on scale.	Ref. 3-g, Appendix C/FDW - p. 5. Indicated level was 9.68 in.
03:03:39 (0604:16)	Turbine bypass valves from steam generator B were isolated by closing block valve MS-V15B.	Ref. 3-g, Appendix STEAM DUMP - pp. 2-4. Steam generator B was isolated.

TIME	EVENT	REMARKS & REFERENCES
03:04:00 (0704:37)	Steam generator B water level was established at approximately 60% on the operating range and was maintained during the next 8.5 hr.	Ref. 1.v, 1.x, Appendix OTSG - pp. 2-3.
03:06:40 (0707:17)	The condensate storage tank B low water level alarm was received. (Nominal alarm setpoint was 28 ft.)	Ref. 3.g, Appendix C/YDW - p. 6. Indicated level was 19.96 ft.
03:10:27 (0711:04)	Emergency feedwater pump 2A (EP-P-2A) was stopped.	Ref. 3.g, 1-p, 1-m. The water levels of steam generators A & B were 60.8% and 62.5%, respectively, on the operating range. All three emergency feed-water pumps were shut down.
03:11:10 (0711:47)	The condenser hotwell low water level alarm cleared. The nominal low level alarm setpoint was 22.5 in.	Ref. 3.g. Indicated level was 23.87 in.
03:12:28 (0713:05) Approximate	The operator opened the ERV block valve (EC-V2). The ERV high outlet temperature alarm was received.	Ref. 3.g, 4.a, 4.d, 4.h, 1.k, 1.g, 2.c - p. 7, 2.m - p. 2, Appendix ERV - pp. 3-5, Appendix TH - p. 63. ERV outlet temperature was 247.7°F. Reactor coolant system pressure and pressurizer coolant level began decreasing, followed by an increase in reactor building pressure.

TIME	EVENT	REMARKS & REFERENCES
03:12:53 (0713:30)	Reactor coolant pump 2B (RC-P-2B) was stopped.	Ref. 3-g. Appendix TH - p. 63. Attempts at forced circulation were again stopped. There had been no indication of flow and motor current had been lost.
03:13:58 (0714:35)	Pressurizer safety valves (R1A & R1B) outlet high temperature alarms were received.	Ref. 3-g.           R1A           202.6°F R1B           202.8°F
03:14:23 (0715:00)	Intermediate Cooling Pump Area Monitor (HP-R-207) radiation reading increased and leveled off at 100 $\mu$ r/hr.	Ref. 12-a, Appendix EH - p. 2.
03:20:23 (0721:00)		
03:17:00 (0717:37)	The ERV block valve (RC-V2) was closed.	Ref. 4-a, 4-h.
Approximate		
03:19:45 (0720:22)	An engineered safety features actuation of high pressure injection was manually initiated.	Ref. 3-g, Appendix EBF - p. 11. Manual initiation is indicated by the computer typewriter entries that groups 1, 2 and 3 of trains A and B have been placed in TEST.

TIME	EVENT	REMARKS & REFERENCES
03:20:13 (0720:50)	Reactor coolant makeup pump 1C (RU-P-1C) started.	Ref. 3.g, Appendix ESP - pp. 11, 14. Pumps 1A & 1C were operating. The reason for the 23 second delay in this pump start following the engineered safety features actuation cannot be explained at this time.
03:20:23 (0721:00) Approximate	All monitors on EP-UR-32f.4 stripchart were showing increasing radiation levels. These monitors are: Primary Coolant Letdown (RU-R-720) both channels, Intermediate Letdown Cooler B (IC-R-1091), Intermediate Letdown Cooler A (IC-R-1092), Intermediate Letdown Cooler Outlet (IC-R-1093), Plant Effluent Unit II, (WEL-R-1311), Decay Heat Closed Loop A (DC-R-3399), Decay Heat Closed Loop B (DC-R-3400), Nuclear Services Closed Cooling (NS-R-3401), Spent Fuel Cooling (SF-R-3402).	Fig. 12.a, Appendix RM - p. 1.
03:21:00 (0721:37) Approximate	Out-of-core nuclear instrumentation indicated sharp decreases in level.	Ref. 4.f, Appendix CI - p. 12. The detector assembly was filled with coolant from high pressure injection, shielding the detector from the core.

TIME	EVENT	REMARKS & REFERENCES
03:21:23 (0722:00) Approximate	Reactor Building Purge Air Exhaust (HP-R-225 & HP-R-226) Auxiliary Building Purge Air Exhaust (HP-R-222) gas radiation monitor readings increased off scale high. The gas channel auxiliary building heating and ventilating radiation monitor was almost off scale. The iodine channel of HP-R-225 & 226 indicated $1 \times 10^5$ cpm.	Ref. 12.a, Appendix EM - p. 3.
03:28:23 (0728:00) Approximate	A general emergency was declared. Notification of off-site authorities was begun.	Ref. 2.b - p. 7, 2.r - p. 3. This action was initiated because of an 8 R/hr radiation reading on the Reactor Building Dome Monitor (HP-R-214).
03:25:56 (0725:33)	The pressurizer high coolant level alarm cleared. (Nominal alarm setpoint was 260 in.)	Ref. 3.g, 1.g, Appendix PDS - pp. 2, 12-13. Indicated level was 238 in. The computer scan interval was 15 sec.
03:27:33 (0729:00) Approximate	The auxiliary building access corridor radiation monitor readings began to increase.	Ref. 12.a, Appendices EM - p. 2, AUX BLDG.
03:29:23 (0730:00) Approximate	The fuel handling building air exhaust fan was turned off. Between 0730 and 1100, this exhaust fan was turned off and on several times with 30 to 60 minute run times.	Ref. 12.a, Appendix EM - p. 3.



TIME	EVENT	REMARKS & REFERENCES
03:30:50 (0731:35)	Pressurizer safety valve (RC-R1A) high outlet temperature alarm cleared.	Ref. 3.g. This indicated a decreasing temperature in the outlet header. Indicated temperature was 192.4°F.
03:32:26 (0733:03)	The pressurizer high coolant level alarm was received.	Ref. 3.g, 1.g. Indicated level was 271 in.
03:34:28 (0735:05)	Pressurizer safety valve (RC-R1B) high outlet temperature alarm cleared.	Ref. 3.g. This indicated decreasing temperature in the outlet header. Indicated temperature was 192.6°F.
03:35:06 (0735:43)	Emergency feedwater pump 2A (EP-P-2A) was started.	Ref. 3.g, 1.v. Steam generator A water level began decreasing from approximately 70% of the operating range and had reached about 43% by 0742.
03:37:00 (0737:37)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.g, 1.g. Pressurizer coolant level indication was increasing rapidly. Indicated level was 374 in.
03:40:00 (0740:37) Approximate	The ERV block valve (RC-V2) was opened.	Ref. 4.a, 4.h.
03:40:28 (0741:05)	Pressurizer safety valves (RC-R1A & RC-R1B) outlet temperatures alarmed high.	Ref. 3.g. Indicated temperatures were 201.6°F and 205.2°F, respectively.

TIME	EVENT	REMARKS & REFERENCES
03-05:23 (0347:00) Approximate	Out-of-core nuclear instrumentation indicated a sharp increase in count rate. Subsequently, the self powered neutron detectors responded to the high temperature conditions over broad sections of the core down to the lowest level.	Ref. 4.f, Appendix CI - p. 12. This change in count rate does not appear to be linked with any operational evolutions associated with the reactor coolant system.
03-09:23 (0349:00) Approximate	Makeup Tank Area Monitor (HP-R-206) radiation reading were increasing off scale. HP-R-210 and HP-R-214 radiation readings leveled out at $1.5 \times 10^2$ R/hr.	Ref. 12.a, Appendix EM - p. 2.
03-55:39 (0356:16)	Engineered safety features actuation system B actuated on reactor building high pressure. The reactor building was isolated by train B valve closures.	Ref. 3.b, 2-a - p. 5, 2-q - p. 4, Appendix ESF - pp. 3, 6, 8, 10, 15 - 18. The nominal setpoint is 4 psig; the trip occurred at 3.2 psig. The actuation of the engineered safety features at this time could be attributed to the opening of the ESF block valve. Letdown flow is stopped whenever the building isolates. It is not possible to verify reinstatement of letdown flow subsequent to building isolations throughout the day. However, it does not appear that letdown flow was reestablished following this isolation.
03-55:39 (0356:16)	Intermediate cooling pump 1B (IC-P-1B) tripped.	Ref. 3.b. This trip was a result of reactor building isolation.

TIME	EVENT	DETAILS & REFERENCES
03:55:40 (0356:17) Approximate	Waste Gas Tank Discharge A Monitor (WOC-P-1485) inflation readings began increasing from 500 cps and reached 3000 cps at 0900.	Ref. H2.a, Appendices H2 - p. 3, H2X H2AC. The monitor was located in the spent resin storage valve room on the 305 ft elevation in the auxiliary building.
03:56:05 (0356:23)	Engineered safety features actuation system A actuated on reactor building high pressure.	Ref. 3.b, 2.a - p. 5, 2.g - p. 4, Appendix H2F - pp. 6, 8. The nominal setpoint is 4 psig; the trip occurred at 3.1 psig.
03:56:05 (0356:23)	Intermediate cooling pump 1A (IC-P-A) tripped.	Ref. 3.b. This trip was a result of reactor building isolation.
03:56:04 (0356:31)	Makeup pump 1C (MU-P-1C) started.	Ref. 3.b. There was an unexplained delay again from the time engineered safety features actuated and makeup pump 1C started.
04:00:00 (0356:37) through	Core thermocouple readings manually obtained indicated core exit temperatures in the range of 217°F to 2500°F.	Ref. 3.b, 2.b - p. 24, Appendix CI - p. 15. It is inferred from the observed temperature pattern that quenching in the core caused breakup of the oxidized fuel cladding and fuel debris redistribution in the core upper regions, blocking or restricting normal coolant flow paths.
05:30:00 (0356:37) Approximate		

TIME	EVENT	REMARKS & REFERENCES
04:00:00 (0000:17) Approximate	Pressurizer coolant level was 100 inches. Reactor coolant pressure was 1500 psig.	Ref. 1.g, 4.d.
04:00:13 (0000:50)	The engineered safety features actuation signal for baffle isolation cleared.	Ref. 1.b, Appendix ESF - p. 7.
04:00:13 (0000:50)	Intermediate cooling pumps 1A and 1B (IC-P-1A & 1B) started. Pump 1A tripped immediately and was restarted.	Ref. 1.b. Flow from the intermediate cooling system is required prior to reactor coolant pump startup.
04:00:17 (0000:54)	Reactor coolant pump 1A (RC-P-1A) was started.	Ref. 1.b.
04:00:18 (0000:55)	Reactor coolant pump 1A (RC-P-1A) was tripped.	Ref. 1.b. There was no flow indication and running current was low.
04:10:10 (0010:47)	Intermediate cooling pump 1B (IC-P-1B) tripped.	Ref. 1.b.
04:17:17 (0017:54)	Subpump pump 1A (SU-P-1A) tripped.	Ref. 1.b, 2.d - p. 10.

TIME	EVENT	REMARKS & REFERENCES
04:17:22 (0817:55)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3-b. No makeup pumps were operating. Coolant injection to the reactor coolant system was completely stopped.
04:18:16 (0818:53)	The operator attempted to restart makeup pump 1A (MU-P-1A).	Ref. 3-b, 2.d - p. 10. The pump would not start. The operator reported locking the pump out at this time.
04:19:02 (0819:39)	Intermediate cooling pump 1B (IC-P-1B) was started.	Ref. 3-b.
04:19:05 (0819:42)	Engineered safety features actuation system A actuated on reactor building high pressure. The reactor building was isolated by train A valve closures.	Ref. 3-b, 4.a, Appendix ESF - pp. 3, 6, 8, 11, 15 - 18. The normal setpoint is 4 psig. The actuation occurred at 3.2 psig. There was no indication of makeup pumps 1A or 1B starting, which could confirm that makeup pump 1A was locked out. There is no explanation for pump 1C not starting.
04:19:06 (0819:43)	Decay heat removal pump 1A (DH-P-1A) started.	Ref. 3-b. This was a result of the engineered safety features actuation.
04:19:06 (0819:43)	Intermediate cooling pump 1A (IC-P-1A) tripped.	Ref. 3-b. This was a result of the engineered safety features actuation.

TIME	EVENT	REMARKS & REFERENCES
04:19:24 (0820:01)	The engineered safety features actuation system A actuation signal cleared.	Ref. 3-b, Appendix ESF - p. 8.
04:19:29 (0820:01)	Intermediate cooling pump 1A (IC-P-1A) was started.	Ref. 3-b.
04:22:02 (0822:39)	Makeup pump 1B (MU-P-1B) was started.	Ref. 3-b, Appendix MCPCS - pp. 2, 3-5.
04:26:22 (0826:59)	Letdown cooler 1A & 1B high temperature alarms were received. (Nominal alarm setpoint was 135°F.)	Ref. 3-b.      1A - 137.7°F 1B - 137.4°F These alarms could be indicative of reinitiation of letdown flow after the engineered safety features actuation signal cleared. The computer scan interval was 30 sec.
04:27:02 (0827:39)	Reactor coolant makeup pump 1C (MU-P-1C) was started. It tripped and was restarted.	Ref. 3-b, 2-d - p. 13.
04:30:30 (0831:07)	Pressurizer heater group 10 tripped.	Ref. 3-b.

TIME	EVENT	REMARKS & REFERENCES
04:30:45 (0831:22)	Condenser vacuum pumps 1A & 1C (VA-P-1A & VA-P-1C) were stopped and main condenser vacuum was broken.	Ref. 3-b, 2-r - p. 4, Problems had been encountered with the auxiliary boiler.
04:35:22 (0835:59)	Both hotdown cooler 1A and 1B high temperature alarms cleared in a 30 sec. period.	Ref. 3-b. Indicated temperatures were 125.9°F and 130.8°F respectively.
04:42:14 (0842:51)	Emergency feedwater pump 2A (EF-P-2A) was shut down.	Ref. 3-b, 1, 1-p, 1-w, Appendix GCSG - pp. 2-3. Water levels in steam generators A and B were 40% and 66% respectively, on the operating range. All three emergency feedwater pumps were shut down.
04:44:23 (0845:00) Approximate	Hotdown Cooler A Monitor (HC-R-1092) radiation readings began decreasing.	Ref. 12-a, Appendix III - p. 2.
04:46:21 (0846:58)	Pressurizer heater groups 4 and 5 tripped off.	Ref. 3-b.
04:59:23 (0900:00) through 05:49:23 (0950:00) Approximate	HP-R-207 and 1B Emergency Cooling Booster Pump Area Monitor (EP-R-204) radiation readings increased to $4 \times 10^3$ mR/hr and then decreased.	Ref. 12-a, Appendix III - p. 2.

## TIME

## EVENT

## REMARKS &amp; REFERENCES

05:18:00  
(0518:37)  
Approximate

The operator closed the ESW block valve.

Ref. 4.a, 4.d, 4.h. This was an apparent attempt to reestablish normal system pressures.

05:18:47  
(0518:24)

Decay heat removal pump 1A (DH-P-1A) was tripped.

Ref. 3.b.

05:20:00  
(0520:37)  
Approximate

Reactor coolant system pressure started to increase from 1250 psig.

Ref. 1.k, 4.d. The increasing trend continued for about 30 minutes until pressure control was established within a band of 1865 to 2150 psig. This increase was a result of the block valve closure.

05:23:34  
(0523:11)

Engineered safety features activation system A actuated on high reactor building pressure. The reactor building was isolated by train A valve closure.

Ref. 3.b, Appendix ESW - pp. 3, 6, 8, 10, 16 - 18. There was no computer typewriter indication of makeup pump 1A (MS-P-1A) starting.

05:23:34  
(0523:11)

Intermediate cooling pump 1A (IC-P-1A) tripped.

Ref. 3.b. This trip resulted from the train A reactor building isolation.

05:23:47  
(0523:24)

The engineered safety features activation system A actuation signal cleared.

Ref. 3.b, Appendix ESW - p. 8.



TIME	EVENT	REMARKS & REFERENCES
05-23-57 (0504-34)	Intermediate cooling pump 1A (IC-P-1A) was started.	Ref. 3.b.
05-29-73 (0530-00) Approximate	Fuel Handling Building Monitor (HF-R-215) and Control and Service Building Corridor Monitor (HF-R-230) radiation readings increased to 40 and 70 $\mu\text{r/hr}$ .	Ref. 12.a, Appendix III - p. 2.
05-29-73 (0530-00) through 14-59-73 (1500-00) Approximate	Radiation readings of the Auxiliary Building Access, Reactor Building Bypass Unit Area, and Fuel Handling Building Exhaust Unit Area monitors all increased to off scale high. HF-R-124 indicated several rapidly increasing and decreasing radiation readings as the trace trended to off scale high.	Ref. 12.a, Appendix III - p. 2.
05-30-57 (0505-34)	HW (HC-22) and pressurizer safety valve (HC-R1A) high outlet temperature alarms cleared.	Ref. 3.b. Indicated temperatures were 192.0°F and 192.9°F, respectively.
05-30-77 (0900-04)	Pressurizer safety valve (HC-R1B) high outlet temperature alarm cleared.	Ref. 3.b. Indicated temperature was 192.9°F.

TIME	EVENT	REMARKS & REFERENCES
05:43:06 (0943:43) Approximate	By cycling the ERV block valve open and closed, approximately 30 times, reactor coolant pressure was maintained between 1865 and 2150 psig during the following two hours.	Ref. 1.k, 4.a, 4.b.
05:43:27 (0944:04) through 05:46:27 (0947:04)	ERV (RC-R2) and pressurizer safety valves (RC-R1A & RC-R1B) outlet temperatures alarmed high.	Ref. 3.b. Indicated temperatures were 214.9°F, 205.9°, and 225.4°F, respectively.
05:49:23 (0950:00) Approximate	Control Room Intake Monitor (HP-R-220) radiation readings increased; the particulate channel reached $2 \times 10^4$ cps, and the iodine channel reached $2 \times 10^3$ cps. By 1100 hours, readings on all channels decreased to below $1 \times 10^2$ cps.	Ref. 12.a, Appendix RM - p. 3.
05:59:23 (1000:00) Approximate	The auxiliary building heating and ventilating exhaust fan tripped.	Ref. 12.a, 2.r - pp. 11 & 12, Appendix RM - p. 3. This was indicated by a decrease of flow on a strip-chart. This trip was reportedly caused by high radiation.

TIME	EVENT	REMARKS & REFERENCES
06:04:00 (1004:37) Approximate	Filling of steam generator A was started.	Ref. 1.p, 1.q, Appendix TH - p. 6B. This could indicate an attempt to establish natural circulation in the reactor coolant system.
06:13:39 (1014:16)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
06:14:06 (1014:43)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
06:14:23 (1015:00) Approximate	The auxiliary building heating and ventilating exhaust fan started.	Ref. 12-a, 2.r - pp. 11 & 12. The high radiation trip was reportedly bypassed.
1100:00 (March 29) through 0100:00 (March 29) Approximate	The fuel handling building air exhaust fan ran steadily.	Ref. 12-a.
07:00:31 (1109:08)	Emergency feedwater pump 2A (EF-P-2A) was started.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
07:14:06 (1114:43)	Indicated steam generator A level reached HMDA on the operating range.	Ref. 1-p.
07:17:01 (1117:38)	Emergency feedwater pump 2A (EF-P-2A) was tripped.	Ref. 3-b.
07:38:54 (1139:31)	The EDV block valve (BC-42) was opened.	Ref. 1.k, 4-a, 4-d, 4.h, Appendix III, p. 74. A rapid sustained depressurization of the reactor coolant system was begun.
07:41:35 (1142:12)	The engineered safety features actuation signals were bypassed.	Ref. 3.b, Appendix EFP - p. 12. This event prevents the engineered safety features of the high pressure injection system from actuating.
07:43:44 (1144:21)	Pressurizer heater groups 1 and 2 tripped off and indicated on 2 seconds later.	Ref. 3-b.
07:44:23 (1145:00) Approximate	The auxiliary building heating and ventilating exhaust fan stopped.	Ref. 12-a.
07:50:16 (1150:53)	Pressurizer heater groups 1 and 2 tripped.	Ref. 3-b.

TIME	EVENT	REMARKS & REFERENCES
08:01:13 (1201:50)	A letdown cooler 1B high temperature alarm was received. (Nominal alarm set point was 135°F.)	Ref. 3.b. The indicated temperature was 140°F. Letdown flow is automatically isolated when this alarm is received.
08:15:22 (1215:59)	A letdown cooler 1A high temperature alarm was received. (Nominal alarm set point was 135°F.)	Ref. 3.b. The indicated temperature was 137.4°F.
08:30:00 (1230:37) Approximate	The power operated emergency main steam dump valve (MS-V3A) was shut at the request of corporate management.	Ref. 2.a - p. 27, 2.d - p. 9, 2.r - p. 4, Appendix STEAM DUMP - pp. 5 - 6. This was in response to concern expressed by the state government.
08:31:06 (1231:43)	Decay heat closed cooling water pumps 1A & 1B (DC-P-1A & DC-P-1B) were started.	Ref. 3.b. This system provides cooling to the decay heat removal system coolers and pumps.
08:31:10 (1231:47) Approximate	Reactor coolant pressure had decreased to 600 psig.	Ref. 3.h, 4.d, Appendices TH - p. 75, CF. Core Flood should have initiated at this time.
08:34:23 (1235:00) Approximate	The auxiliary building heating and ventilating exhaust fan started momentarily, then remained off for about one and a quarter hours.	Ref. 12.a.
09:04:18 (1304:55)	Makeup pump 1C (MU-P-1C) stopped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
09:10:00 (1310:30) Approximate	ERV block valve (RC-V2) was closed. Reactor coolant system pressure had decreased to approximately 435 psig and then began to increase.	Ref. 4.a, 4.d, 4.h, Appendices TH - p. 75, CF. Core flood apparently stopped after approximately 100 ft <sup>3</sup> (770 gal) of coolant had been injected into the reactor coolant system from each of the two core flood tanks.
09:16:58 (1317:35)	ERV (RC-822) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 192.7°F.
09:19:52 (1320:25)	The hydrogen cooler 1A high temperature alarm cleared.	Ref. 3.b. The indicated temperature was 131.3°F.
09:20:28 (1321:05)	ERV (RC-822) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 220.4°F. The ERV block valve may have been opened.
09:31:28 (1332:05)	ERV (RC-822) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 192.2°F. The ERV block valve may have been closed.
09:48:58 (1349:35)	ERV (RC-822) outlet temperature alarmed high.	Ref. 3.h. The indicated temperature was 225.7°F. It had been reported that the ERV block valve was opened at the time of the hydrogen burn.

TIME	EVENT	REMARKS & REFERENCES												
09:49:23 (1350:00) Approximate	The auxiliary building heating and ventilating exhaust fan was started and ran for about 30 minutes.	Ref. 12-a.												
09:49:43 (1350:20) through 09:49:50 (1350:27)	Engineered safety features high pressure injection, reactor building isolation, reactor building spray pumps and valves, and decay heat removal pumps were actuated. Makeup pump (MU-P-1C) started. There was no indication of makeup pump 1A (MU-P-1A) running or starting.	Ref. 3-b, 4.a, 2-a - p. 22, 2.1 - p. 4, 2-4 - p. 8, 2.5 - p. 1, Appendix ESP - pp. 6, 8 - 10. Actuation was caused by reactor building high pressure which was caused by a hydrogen burn.												
09:49:43 (1350:20) Approximate	Motor control centers 32A & 42A were lost.	Ref. 2-a - p. 24, 2-x - p. 7, 2-q - p. 8. These motor control centers supplied all the auxiliaries for the four reactor coolant pumps and seal water pumps for many of the pumps in the auxiliary building. This event coincided with the hydrogen burn.												
09:49:50 (1350:35)	Reactor coolant pumps 1A and 1B (RC-P-1A & RC-P-1B) inlet air temperature high alarms were received, and pressurizer safety valves (R1A & R1B) discharge line temperature high alarms annunciated.	<table border="1"> <tr> <td data-bbox="1220 987 1339 1009">Ref. 3-b.</td> <td data-bbox="1436 987 1535 1009">RC-P-1A</td> <td data-bbox="1619 987 1717 1009">157.5°F</td> </tr> <tr> <td></td> <td data-bbox="1436 1030 1535 1052">RC-P-1B</td> <td data-bbox="1619 1030 1717 1052">124.7°F</td> </tr> <tr> <td></td> <td data-bbox="1436 1073 1482 1095">R1A</td> <td data-bbox="1604 1073 1703 1095">203.7°F</td> </tr> <tr> <td></td> <td data-bbox="1436 1116 1482 1138">R1B</td> <td data-bbox="1604 1116 1703 1138">205.0°F</td> </tr> </table> <p data-bbox="1220 1158 1906 1220">These temperature alarms could have been caused by the hydrogen burn.</p>	Ref. 3-b.	RC-P-1A	157.5°F		RC-P-1B	124.7°F		R1A	203.7°F		R1B	205.0°F
Ref. 3-b.	RC-P-1A	157.5°F												
	RC-P-1B	124.7°F												
	R1A	203.7°F												
	R1B	205.0°F												

TIME	EVENT	REMARKS & REFERENCES
09:50:11 (1350:48)	The reactor building isolation and cooling actuation signal was defeated.	Ref. 3.b, Appendix ESE - pp. 12-13. Shutdown of the makeup pumps operating in the high pressure injection mode is made possible by defeating the engineered safety system actuation signal.
09:50:11 (1350:48)	Intermediate cooling pumps 1A and 1B (IC-P-1A & IC-P-1B) were started.	Ref. 3.b.
09:50:24 (1351:01)	Makeup pump 1C (MU-P-1C) was stopped.	Ref. 3.b.
09:51:58 (1352:35)	The ERV (RC-R2) and pressurizer safety valves (RC-R1A) discharge line high temperature alarms cleared.	Ref. 3.b.      ER2      180.5°F ER1A      178.6°F The ERV block valve may have been closed.
09:52:28 (1353:05)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 208.8°F. The ERV block valve may have been opened.
09:54:28 (1355:05)	Pressurizer safety valve (RC-R1B) high outlet alarm cleared.	Ref. 3.b. The indicated temperature was 177.3°F. The ERV block valve may have been closed.
09:55:10 (1355:47)	Pressurizer heater group 8 tripped.	Ref. 3.b.



TIME	EVENT	REMARKS & REFERENCES
09:55:28 (1356:05)	Pressurizer safety valve (RC-R1B) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature alarm was 190.3°F.
09:55:30 (1356:07)	Reactor building spray pumps were stopped.	Ref. 3.b, Appendix KSF - pp. 12-13.
09:56:58 (1357:35)	Decay heat removal pumps 1A & 1B (DB-P-1A & DB-P-1B) were stopped.	Ref. 3.b, Appendix CF.
09:58:38 (1359:15)	The engineered safety features actuation signal cleared.	Ref. 3.b, Appendix KSF - p. 8.
10:00:08 (1400:37)	The KW block valve (RC-W2) was opened.	Ref. 4.a, 4.b, Appendix 2B - p. 76.
Approximate		
10:00:58 (1401:35)	KW (RC-W2) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 286.7°F.
10:05:25 (1406:02)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
10:07:19 (1407:56)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
10:26:15 (1426:52)	The reactor coolant system loop A hot leg temperature decreased to less than 620°F.	Ref. 1.b, 2.s - p. 1.
10:31:30 (1432:07)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b. The operation of this pump during the following hour could be indicative of efforts to maintain or increase the coolant inventory in the reactor coolant system.
10:32:36 (1433:13)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
10:34:00 (1434:37) Approximate	Reactor coolant pressure decreased again to approximately 435 psig.	Ref. 4.d, Appendices TH - p. 76, CF. Core flood should have initiated again.
10:35:00 (1435:37) Approximate	Reactor coolant pressure had decreased to approximately 410 psig and began to increase.	Ref. 4.b, Appendices TH - p. 76, CF. Core flood injection apparently stopped. An additional estimated 22 ft <sup>3</sup> (161 gal) of coolant had been injected into the reactor coolant system from each flood tank.
10:35:55 (1436:32)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
10:38:57 (1439:34)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
10:39:51 (1440:28)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
11:08:00 (1508:37) Approximate	The EHV block valve (RC-V2) was closed.	Ref. 3.b., 4.a, 4.b.
11:18:34 (1519:11)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b.
11:24:29 (1525:06)	EHV (RC-V2) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 191.9°F.
11:28:12 (1528:49)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.
11:28:52 (1529:29)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
11:32:37 (1533:14)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
11:33:44 (1534:21)	Emergency feedwater pump 2B (EF-P-2B) was started.	Ref. 3.b.
11:35:48 (1536:25)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.
11:38:57 (1539:34)	Feedwater flow to steam generator B was started.	Ref. 1.v.
	Approximate	
14:45:17 (1545:54)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
11:52:08 (1552:41)	Emergency feedwater pump 2B (EF-P-2B) was tripped. Feedwater flow to steam generator B was stopped.	Ref. 3.b., 1.m.
12:34:00 (1636:37)	ERV block valve (RC-V2) was opened.	Ref. 3.h, 4.a, 4.b.
	Approximate	
12:34:29 (1635:05)	ERV (RC-E2) outlet temperature alarmed high.	Ref. 3.b. Indicated temperature was 233°F.

TIME	EVENT	REMARKS & REFERENCES
12-46-68 (1646-37) Appropriate	ERV block valve (RC-V2) was closed.	Ref. 3.h, 4.a, 4.h.
12-52-68 (1652-37) Appropriate	ERV block valve (RC-V2) was opened.	Ref. 4.a, 4.h.
13-08-68 (1708-37) Appropriate	ERV block valve (RC-V2) was closed.	Ref. 4.a, 4.h.
13-02-23 (1702-09)	Condenser vacuum pump 1C was started.	Ref. 3.h. This event and the subsequent one indicate the reestablishment of vacuum in the main condenser.
13-13-68 (1713-47)	Condenser vacuum pump 1A was started.	Ref. 3.h.
13-23-68 (1723-41)	Makeup pump 1C (MU-P-1C) started.	Ref. 3.h.
13-28-68 (1728-35)	ERV (RC-B2) high outlet temperature alarm cleared.	Ref. 3.h. Indicated temperature was 192.9°F.

TIME	EVENT	REMARKS & REFERENCES
13-25-09 (1325-46)	Pressurizer heater groups 1 and 2 tripped.	Ref. 3.b.
14-25-26 (1425-03)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
14-43-15 (1443-52)	Makeup pump 1C (MC-P-1C) tripped.	Ref. 3.b.
14-55-35 (1455-12)	Reactor coolant system pressure was 2127 psig.	Ref. 3.b.
14-58-23 (1458-00) Approximate	The Fuel Handling Building Exhaust Unit Area Monitor, Reactor Building Purge Unit Area Monitor, and Auxiliary Building Access Area Monitor radiation readings came back on scale and were decreasing.	Ref. 12.a, Appendix III - p. 2.
15-12-42 (1512-19) Approximate	Reactor coolant pump 1A (RC-P-1A) was started and ran briefly, then stopped.	Ref. 3.b, 1.f, 2.a - p. 25. There was still no power available from motor control centers 32A and 42A for reactor coolant pump auxiliaries. Necessary circuit bypasses were installed to permit operation of the pump.

TIME	EVENT	REMARKS & REFERENCES
15:49:36 (1950:13) Approximate	Reactor coolant pump 1A (RC-P-1A) was restarted.	Ref. 3.h, L.F., 2-a - p. 25, 2.s - p. 4. Flow was indicated at 1950:46 on the manometer.
16:29:23 (2030:00) Approximate	Radiation readings showed marked increases and went off or nearly off scale on the Fuel Handling Building Exhaust Shift Area Monitor (HP-R-3236), Auxiliary Building Access Corridor Monitor <sup>3</sup> (EP-R-232), Waste Disposal Storage Area Monitor (HP-R-208).	Ref. 12.a, Appendix RM - p. 2.

Plant Status at 2000:00

At approximately 2000:00 on March 28 (16 hours after the initiating event, indications show that forced circulation had been reestablished using reactor coolant pump 1a (RC-P-1A). The reactor coolant system pressure was being maintained at approximately 1800 to 1900 psig with temperatures indicating a cooling trend. Heat was being removed from the reactor coolant system using steam generator A. Steam generator B was isolated and condenser vacuum had been established.

Holston, Coleda | Thursday, February 9, 2012

Document:  **"Analysis of TMI-Unit 2 Accident."** (Version 1.0, Released)

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## Class Properties

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ADAMS Permanent Version?	No
ADAMS Checkin Date	May 24, 2001
ADAMS Added By User	Legacy394
ADAMS Archived?	No

431-97-025  
Box 7



ADAMS Date Added            May 24, 2001  
ADAMS Checked In By User   Legacy394

System Properties

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Modified On:	7/31/11 9:47 AM
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Is Current Version:	True
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Minor Version:	0
Version Status:	Released
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Buckley, John | Friday, February 10, 2012

Document:  **"Analysis of TMI-Unit 2 Accident."** (Version 1.0, Released)

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## Class Properties

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FACA Document	No
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ADAMS Archived?	No
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ADAMS Checked In By User	Legacy394

## System Properties

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Minor Version:	0
Version Status:	Released
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Buckley, John | Friday, February 10, 2012

Document:  **"TMI-2 Post-Defueling Survey Rept for RCS."** (Version 1.0, Released)

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#### Class Properties

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ADAMS Archived?	No
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97-0042  
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System Properties

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TMI-2 POST-DEFUELING SURVEY REPORT


FOR

THE REACTOR COOLANT SYSTEM

Approved:

  
\_\_\_\_\_  
Manager, TMI-2 Licensing

Approved:

  
\_\_\_\_\_  
Manager, TMI-2 Engineering

TMI-2 INTERNAL REVIEWS

Authors:

*P. W. Deininger* 6-27-91  
P. W. Deininger

Responsible Technical Reviewer:

*R. H. Freeman* 6-27-91  
R. H. Freeman

Manager, Fuel Measurements & Analysis:

*C. H. Distenfeld* 6-27-91  
C. H. Distenfeld

PDSR Committee Chairman:

*R. E. Rogan* 6-28-91  
R. E. Rogan

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TMI-2 POST-DEFUELING SURVEY REPORT  
FOR THE REACTOR COOLANT SYSTEM

SUMMARY

The estimate of record of the amount of uranium dioxide ( $UO_2$ ) remaining in the RCS Components included in this Post Defueling Survey Report (PDSR) was 25.8 kg with an uncertainty of  $\pm 43\%$ , distributed as follows:

Cold Legs '1A/B' & '2A/B'	9.6 kg $UO_2$
Reactor Coolant Pumps	6.2 kg $UO_2$
Surface Films	4.6 kg $UO_2$
Hot Legs 'A' & 'B'	2.7 kg $UO_2$
Decay Heat Line	1.5 kg $UO_2$
Core Flood Lines 'A' & 'B'	1.0 kg $UO_2$
Pressurizer Lines	<u>0.2 kg <math>UO_2</math></u>
Total	25.8 kg $UO_2$

The above summary table shows that 61% of the  $UO_2$  remaining in the RCS Components was located in two (2) categories of components, the Cold Legs and the Reactor Coolant Pumps. The estimates of 9.6 kg  $UO_2$  in the Cold Legs and 6.2 kg  $UO_2$  in the Reactor Coolant Pumps are based on the observations of reactor fuel debris during video examinations which is assumed to have settled from the static volume of coolant containing a uniform dispersion of particles.

The fuel content of each RCS Component was determined using gross gamma-ray surveys, engineering analyses and video inspections. These methods are described

in Section 3.0. Several RCS components not addressed in this PDSR are discussed in other PDSRs as noted in Section 1.0, Introduction.

In addition, 4.6 kg of fuel ( $UO_2$ ) was estimated to be present in the form of surface films throughout the RCS. This assumes that the RCS surface film deposition pattern was uniform and that the composition was similar to that of the materials present on the components inspected. The error assigned for the surface film estimates was  $\pm 60\%$  (assumed to be one sigma).

The term "uncertainty" in this Post Defueling Survey Report was used to represent the estimated error of each "estimate of record" and was taken as one sigma. The overall uncertainty for the total  $UO_2$  remaining in the RCS Components was estimated as  $\pm 43\%$  based on the square root of the sum of the squares of the uncertainty values for each component.

TMI-2 POST-DEFUELING SURVEY REPORT  
FOR THE REACTOR COOLANT SYSTEM

1.0 INTRODUCTION

This report presents an analysis of the amount of fuel ( $UO_2$ ) remaining in the Reactor Coolant System (RCS). The content of this analysis addresses the fuel remaining in the following components/systems.

- a. Reactor Coolant Pumps RC-P-1A/B and RC-P-2A/B
- b. Core flood lines "A" and "B" from the Reactor Vessel to the CF-V5A/B valves
- c. Cold Legs 1A/B and 2A/B
- d. Hot Legs A and B
- e. Decay Heat Drop Line
- f. Pressurizer Lines
- g. Surface Films

The remaining components of the RCS are addressed in other Post-Defueling Survey Reports as listed below:

<u>RCS Component</u>	<u>Addressed in PDSR</u>
Core Flood Tanks & Piping	RB Miscellaneous
Cold Leg Nozzles	Reactor Vessel
Hot Leg Nozzles	Reactor Vessel
OTSGs	OTSGs
Pressurizer	Pressurizer
RC Drain Tank	RB Miscellaneous
Reactor Vessel	Reactor Vessel
Letdown Coolers	Letdown Cooler Room

This report is one in a series of reports prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 1). All

statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated. Other segments of the Reactor Building (RB) are treated in separate Post-Defueling Survey Reports (PDSR).

Section 2.0, "Background", describes the physical attributes of the individual components/systems and their relationship to the accident and subsequent cleanup activities. The boundaries of these components/systems with respect to this PDSR are also presented in this section.

Section 3.0, "Methods", describes how existing video inspections, sample data and gamma spectroscopy data, where available, were used to determine the estimate of record.

Section 4.0, "Analysis", explains how the estimate of record of fuel in the RCS was determined based on video inspections and sample analysis, and gamma spectroscopy data which was collected for selected RCS components/segments reported herein. This section also discusses supporting data, assumptions made and calculations used.

Section 5.0, "Conclusion", presents the estimate of record and associated uncertainty for each RCS component/segment and reports these values in terms of remaining quantities of  $UO_2$  (in kg) for each RCS segment. Additional rationale is presented leading to the conclusion that the estimate of record is reasonable based on the available data and analyses performed.

## 2.0 BACKGROUND

The March 1979 TMI-2 accident resulted in significant damage to the reactor core and subsequent release of fuel particles and fission products into the Reactor Coolant System (RCS) and other closely interconnected systems/components. The damaged core consisted of loose fuel pellets, solidified fuel, structural metal components, loose rubble, and partial fuel assemblies. Therefore, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

During the accident, core debris was transported to the reactor building (RB) as a result of the core degradation event and coolant flow from the reactor vessel (RV) through the pilot operated relief valve (PORV) and the RCS Makeup and Purification (MU&P) System. Less than 2 kg of fuel ( $UO_2$ ) was transported to the RB during the accident sequence.

The RCS was designed to transfer thermal energy from the reactor core to the steam generators (OTSG's). The RCS also provides neutron moderation and a secondary boundary for preventing the release of fission products to the environment. The RCS consists of the reactor vessel, two (2) vertical OTSG's, four (4) shaft sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. Only those sections of the RCS presented in Section 1.0 are addressed in this report. In addition, two (2) pipe sections of the Core Flood System and the Decay Heat Drop Line are addressed.

### 2.1 Reactor Coolant Pumps

Four (4) reactor coolant pumps are connected to the RCS. The pumps are vertical suction, horizontal discharge and single stage centrifugal units manufactured by Bingham Pump & Williamette, Inc. The pumps are electrically driven by 9000 HP Allis Chalmers,

squirrel-cage induction motors. The pumps and their locations are shown in Figures 1 and 2. All four (4) pumps are located within the concrete D-ring sections of the Unit 2 Reactor Building at approximately elevation 320 feet. Since the top of the D-rings is at elevation 367 feet, the reactor coolant pumps are about forty-seven (47) feet down into the D-rings.

The reactor coolant pumps were operating during the initial phase of the TMI-2 accident. The plant was operating at about 97% power. At about seventy-three (73) minutes into the accident, the 1B and 2B reactor coolant pumps were turned off to reduce the possibility of serious pump damage due to low system pressure. The 1A and 2A pumps continued to circulate coolant through the system until about one hundred (100) minutes into the accident when they also were shut down. The operators attempted to restart the reactor coolant pumps with little success during the next several hours. Reactor coolant pump 2B was started at about three (3) hours into the accident and ran briefly before being shut down. Twelve (12) hours later reactor coolant pump 1A ran for about one (1) minute and was shut down again, and then restarted (See Figure 3). Core debris was circulated through the four (4) reactor coolant pumps during their subsequent operation (Reference 2).

For purposes of this report, the RCP boundaries are that section of the Reactor coolant pumps between the discharge and suction nozzles (shown in Figure 2).

## 2.2 Core Flood Lines "A" and "B"

The core flood system consists of two (2) flood tanks containing approximately 7800 gallons of water (2/3 full), each connected to

the reactor vessel via fourteen (14)-inch piping. Each piping system contains at least two (2) check valves and one (1) stop valve. The system activates automatically as the RCS pressure drops below 600 psig due to a 600 psig nitrogen over pressure in each tank. Only two (2) sections of the core flood system piping (Figure 4 & 5) between the reactor vessel and the CF-V5A and 5B valves are addressed in this report.

The remaining parts of the core flooding system are covered in the Reactor Building Miscellaneous Components PDSR.

During the accident, the core flood system activated as observed by the reduced nitrogen overpressures. During subsequent repressurization of the RCS, back pressure may have forced fuel debris into the core flood piping on the "A" side (Reference 2). Fuel debris has been detected beyond the CF-V5B valve and has been observed in the piping between the reactor vessel and the CF-V5A/B valves (Reference 3).

### 2.3 Cold Legs 1A/B and 2A/B

The cold leg discharge piping sections are located at the elevation 315 feet between the reactor coolant pumps and the reactor vessel. All four (4) pipe sections are twenty eight (28)-inches in diameter and are approximately twenty five (25)-feet in length. RCS coolant flow deposited fuel debris in these pipe sections (Figure 6).

### 2.4 Hot Legs A and B

The "A" and "B" hot legs are horizontal sections of the thirty six

(36)-inch RCS piping that extend east and west from the reactor vessel at about elevation 315 feet. Each hot leg has a horizontal section about twelve (12) feet in length that turns upward (90°) and reaches the 360 foot elevation at the top of the OTSG's (Figure 7). Coolant flow exiting the reactor vessel deposited fuel debris in the twelve (12) foot horizontal section of these two (2) pipe segments.

## 2.5 Decay Heat Drop Line

The Decay Heat Drop Line is a twelve (12)-inch diameter pipe section that attaches to the "B" hot leg at the 90° elbow (Figure 7 & 8). It extends downward to about elevation 291' and then north for approximately eighteen (18) feet before turning west.

The purpose of the Decay Heat System is to remove decay heat from the core and residual heat from the reactor coolant system during shutdown. During the accident, RCS coolant exiting the reactor vessel into the thirty six (36)-inch "B" hot leg moved past the entrance of the decay heat line and deposited fuel debris into the vertical decay heat piping segment. The Decay Heat System was not used during the accident so that the material deposited in the decay heat piping was due to gravitational settling from the "B" hot leg. For purposes of this report, the fuel debris was measured near the elbow of the decay heat line at elevation 291 feet where the decay heat line turns north toward the Fuel Handling Building.

## 2.6 RCS Nozzles

The RCS Nozzles are tapered sections of the reactor vessel which form the entrance to the cold and hot leg piping sections. The core



flood nozzles are not considered in this section but are contained within the estimated fuel debris values for the core flood piping system (Section 4.2). The hot leg nozzles are assumed to contain no fuel debris because of the fuel removal efforts, which effectively removed the fuel debris deposits. The cold leg nozzles are considered separately in the PDSR covering the dry Reactor Vessel fuel deposition.

## 2.7 Pressurizer Lines

The pressurizer upper head has three (3) nozzles connected to the safety relief system and one (1) nozzle connected to the spray line. It also has a vent nozzle and a sixteen (16)-inch internal diameter manway opening. The lower head has a ten (10)-inch nozzle connected to the surge line. A surge diffuser is centered in this nozzle, extending up into the lower head a distance of approximately one (1) foot. The surge line is a ten (10)-inch stainless steel pipe and is located inside the A D-ring. It runs from the vertical portion of the reactor coolant hot leg to the bottom of the pressurizer. Fuel debris was measured in the horizontal section of the surge line.

The Power Operated Relief Valve (PORV) stuck open early in the accident resulting in coolant flow through the pressurizer surge line into the pressurizer and out through the relief valve. This flow was terminated later in the accident by closing the block valve upstream of the PORV. Subsequent to the core damage, this block valve was cycled open for significant periods of time, reestablishing flow through the surge line. The varying fluid levels in the pressurizer and the varying flow rates through the surge line are potential separation mechanisms by which fuel fines

entrained in the fluid could be deposited in the surge line. Since the L-shaped section of the surge line is much lower than its exit point, a significant rate of flow would be required to transport fuel out.

Fuel debris in the pressurizer spray line was flushed back into the pressurizer and was subsequently removed during defueling operations. Therefore, it is assumed that no measurable quantity of fuel remains in the pressurizer spray line.

## 2.8 Surface Films

In addition to the concentrations of residual core debris, there was some fuel bound to the surfaces of the components in the RCS. These deposits were not distinguishable by video inspection because they were thin and relatively uniform over a wide area. This film accounts for a very small fraction of the total fuel left in the RCS, but is included in this report for completeness.

Fuel and related materials on the inside surfaces of the RCS, the pressurizer and OTSG manway and inspection port cover plates were inspected, beta/gamma surveyed, alpha counted and sampled (Reference 22). Subsequently, these plates were counted with a gamma-ray spectrometer and the scrape samples were analyzed for fuel isotopes. The film on each plate differed markedly in thickness. The pressurizer had the thinnest film which was semi-transparent and resembled gun metal blue. The 'B' OTSG had the thickest film which resembled a thick coat of flat black paint, with a tint of blue/brown. All of the films were very hard, uniform and adherent. It appeared that the films originated from both corrosion and deposition; the thicker films appeared to be mostly deposition. The

amount of fuel deposited was roughly proportional to the film thickness, but evidence suggests that much of the fuel is on the surface of the film.

## 2.9 Radiation Environment and Component Access

All of the segments of the RCS described under the Introduction Section (Section 1.0) of this report are located in regions of the D-rings or are contained between the reactor vessel and the D-ring openings. In the case of the cold and hot leg sections, these components penetrate the primary shield wall of the reactor vessel and then extend into the D-rings at about elevation 315 feet. Personnel access to these piping segments would require extended worker exposure times in radiation fields of greater than 1 R/hr in the "A" D-ring (Reference 4, 5, and 6). No prior access to the 315 foot elevation in the "B" D-ring has been permitted because exposure rates range from 1 to about 10 R/hr (Figure 9 and References 6, 7 and 8). At the decay heat drop line elbow (elevation 291 feet), the exposure rate is about 16 R/hr. Therefore, personnel access is not warranted or consistent with ALARA.

Because of these elevated exposure rate environments, all sampling and video inspections were performed from the work platform through the reactor vessel or from a higher elevation of the D-rings (above the 350 foot elevation). Access through the reactor vessel via the defueling work platform was performed remotely, with long handled tools and/or equipment of special design. This allowed personnel to work in a 10 to 25 mR/hr (Reference 6) exposure rate environment.

## 3.0 METHODS

### 3.1 Reactor Coolant Pumps

The estimate of record of the amount of fuel ( $UO_2$ ) remaining in the Reactor Coolant Pumps RC-P-1A/B and RC-P-2A/B was obtained by engineering analysis and review of video inspection tapes (Reference 10, 11, and 12).

The first estimate was produced using a video camera inserted from the defueling work platform through the discharge pipes (cold legs) and into the pump casings (Reference 11). The debris estimate is related to the pump internal cavity volume between the discharge and intake nozzles. Close examination of the video tapes provided sufficient evidence to ascribe an average sediment depth of 1/4 inch (6.4 mm) inside RC-P-2A, and the total volume was estimated to be 11.7 liters of material for this pump (Reference 10). An attempt to remove a sample from the RC-P-1A pump casing failed.

In an effort to apply some reasonable limits to the sediment volume in each pump, it was decided to weight the RC-P-1A, 1B and 2B pump sediment volumes by the proportional fuel debris quantity that was measured in each of their related discharge pipes (cold legs) (Reference 12). This method established the RC-P-2A and the 2A cold leg as the reference or standard. The 2A cold leg was measured as containing 3.9 kg of  $UO_2$  distributed in 21.6 liters of debris (Reference 12) and the RC-P-2A pump casing was estimated to contain 11.7 liters of sediment (Reference 12).

The conversion from volume of sediment to fuel ( $UO_2$ ) was performed by using the results of the sample analysis of materials collected from the cold legs as recorded in Reference 12. The analysis

indicated that the cold leg material density was approximately 3.2 grams per cubic centimeter (Reference 12). The  $UO_2$  fraction was assumed to be 0.58 (Reference 13). These values were used to calculate the final estimate of record for the Reactor Coolant pumps.

### 3.2 Core Flood Lines "A" and "B"

The estimate of record for the core flood lines (from the RV to the CF-V5A and 5B valves) was based on the results produced from gross gamma-ray surveys internal to the Core Flood piping system. The Microshield computer code (Reference 14) was used to model the source geometry established from video inspections of the core flood line deposition patterns. The measured exposure rate was divided by the calculated exposure rate per volume of fuel ( $UO_2$ ), which provided the quantity of fuel remaining in the core flood piping (Reference 12).

The quantity of fuel remaining in the core flood system is assumed to be represented by the values developed for a measured eight (8)-foot segment of both core flood lines ("A" and "B"). No fuel debris was assumed to have reached the CF-V5A valve in the "A" core flood side. On the "B" side, fuel deposits were measured at the CF-V5B valve and in the small drain line that drops from the core flood line near the CF-V5B valve (Reference 3).

### 3.3 Cold Legs 1A/1B and 2A/2B

The cold leg sections extending to the reactor coolant pumps are twenty eight (28) inches in diameter and are approximately

twenty five (25) feet long. The gross gamma-ray measurement plan for each cold leg was compared with the computer model developed from the video inspection of each cold leg and sample data (Reference 12). The measurement method was the same as described for the core flood lines, Section 3.2.

#### 3.4 Hot Legs "A" and "B"

The estimate of record for the hot leg sections was based on gross gamma-ray results produced from internal surveys of the thirty six (36)-inch hot leg sections (Reference 12). As in Section 3.2 above, a computer model was used to compare the values measured by the GM detectors to the fuel quantity in each hot leg. The measurements were performed after the hot legs were cleared of as much fuel debris as possible using specially developed tooling (Reference 15 and 16).

#### 3.5 Decay Heat Drop Line

The decay heat drop line is located below the elbow of the "B" hot leg where the hot leg turns upward (90°) to eventually meet the top of the "B" OTSG. The decay heat line was inspected with a video camera system. From the depth of insertion it was determined where the fuel debris was located at the lower elbow. Defueling was performed by using a vacuum tool mounted on a steel bar with a video camera on one side of the suction nozzle and a light on the other side. The steel bar was designed to maintain the vacuum nozzle as close as possible to the center of the decay heat drop line. Removal of the material was halted when the vacuum system could not penetrate a solidified mass of material within the pipe (Reference

15, 16, 17, and 18). The solidified mass of material was removed in early 1989 using a drain cleaning machine to break up the hard debris. Vacuuming continued and the rest of the loose debris was removed. A GM detector system, similar to those used in the cold leg measurements, was inserted into the decay heat drop line after the debris was removed (Reference 19). The GM measurements were taken over the first three (3) feet of the horizontal section of the decay heat drop line beyond the lower elbow where it turns north for about eighteen (18) feet of horizontal run. The measured values over the first three (3) foot length of pipe were extrapolated over the entire horizontal pipe length.

Measurement values obtained from the gross gamma-ray measurements internal to the decay heat drop line were correlated to fuel debris quantities determined from decay heat drop line sample results (Reference 20). Those results indicate that the material collected was about 8%  $UO_2$  by weight.

### 3.6 Reactor Coolant Nozzles

The cold leg nozzles are about seventeen (17) inches long and form the entrance to the twenty eight (28)-inch cold leg pipe sections. They form a tapered entrance of approximately 15°; starting at a diameter of approximately thirty seven (37) inches. Video inspections of three of four nozzles indicated that a sediment layer remained in the flanged sections. Estimates of the total debris remaining are included in the Dry Reactor Vessel Post Defueling Survey Report. No fuel movement in these regions is anticipated except during water flow, and the remaining quantities of fuel debris will generally remain in their present locations until TMI-2 is decommissioned except for the fuel located in the '1B' nozzle.

During the RV fuel measurement sequence, it will be necessary to drain the RV while making passive neutron measurements. During the refill cycle, the RV will be filled using water from the RCBTs via the "B" makeup line (MU-V-16B). This filling operation is expected to wash most of the residual fuel out of the "1B" cold leg and nozzle into the RV.

### 3.7 Surface Films

Three (3) different methods were employed to measure the film deposits on various RCS components (Reference 9). Beta/gamma exposure rates were measured using an ion chamber type survey meter (Eberline Model RO2A) and the alpha count rates were measured with a thin window proportional counter (Eberline Model PAC-6). The beta exposure rates were used to estimate the Sr-90 activity and the alpha count rates were used to estimate fuel deposition. The manway plates were also counted with a collimated HPGe gamma-ray spectrometer using Ce/Pr-144 and Eu-154 as analogs for fuel estimates. The film deposits were estimated using the three (3) independent measurements: the HPGe spectrometer (Ce/Pr and Eu lines), the direct alpha count rate, and the radiochemistry of the scrape samples. The radiochemistry results directly measured the fuel isotopes and are assumed to be most accurate. The collection efficiency (50%) was estimated visually and by alpha counting before and after scraping. It includes the fraction of the area cleaned and the fraction picked up. Therefore, the accuracy of the scrape sample is controlled by the collection efficiency (i.e., no worse than 50%).



### 3.8 Pressurizer Lines

The only Pressurizer line that was surveyed was the surge line. A portable directional survey instrument (Eberline Model HP-220A) was used to measure the exposure rates at discrete locations along the horizontal section of the surge line. Higher intensities were noted along the eastward section of the surge line. Directional measurements along the north-south surge line section did not suggest the presence of additional source material. The impact of residual fuel from the pressurizer on these measurements was reduced by use of a 5-centimeter (2-inch) thick lead block shield (Reference 21).

## 4.0 ANALYSIS

### 4.1 Reactor Coolant Pumps (RCP's)

Video inspection of the Reactor Coolant Pump RC-P-2A casing and discharge leg indicated a sediment layer existed in both. The sediment volumes were evaluated for RC-P-2A and for the 2A cold leg and estimated to be 11.7 and 21.6 liters, respectively (Reference 11 and 12). Direct gross gamma measurements were used to infer the amount of residual fuel in the cold legs (Reference 12). The quantity of fuel in the 2A cold leg was estimated as 3.9 kg UO<sub>2</sub> (Reference 12). The measurement method was unable to access the pump casing. The fuel estimates for the pumps were based on the cold leg measurements scaled by the ratio of the observed sediment volumes in the pump and its cold leg. For the 2A pump per Reference 12:

$$\text{Pump (2A)} = \frac{11.7 \text{ liters (pump)}}{21.6 \text{ liters (cold leg)}} \times 3.9 \text{ kg UO}_2 \text{ (cold leg)} = 2.1 \text{ kg UO}_2$$

It was not possible to inspect the top of the volute, but its volume was scaled from the drawings and is included in the total pump volume (Reference 12).

The total fuel content for RC-P-2A and discharge leg was estimated to be 2.1 kg UO<sub>2</sub>. From the Cold Leg analysis performed in Reference 12, it was determined that the Cold Leg "2A" contained most of the fuel debris. On this basis the remaining RC pumps were prorated using the fuel content of their respective cold legs (Reference 12), and the results were as follows:

RCP-1A contained	1.8 kg UO <sub>2</sub>
RCP-1B contained	1.0 kg UO <sub>2</sub>
RCP-2A contained	2.1 kg UO <sub>2</sub>
RCP-2B contained	<u>1.3 kg UO<sub>2</sub></u>
Total fuel	6.2 kg UO <sub>2</sub>

The uncertainty was determined using the square root of the sum of the squares of the individual uncertainties. The uncertainty was determined to be  $\pm 55\%$  overall for the hot legs, cold legs, RC pumps and CF lines grouped together (Reference 12) and is reflected in Table 2.

#### 4.2 Core Flood Lines "A" and "B"

The fuel content remaining in the Core Flood Lines was determined using gross gamma-ray measurements in conjunction with the Microshield computer code (Reference 14). A one (1)-foot segment of fuel debris on the inside surface of the core flood line piping was modeled on the computer based on a rectangular solid source distribution of RV lower head sample material. Since most of the exposure rate from the deposited material was generated by Cs-137, the cesium-to-fuel ratio and source geometry were the most critical parameters.

The GM detectors were positioned about two (2) centimeters from the bottom of the Core Flood pipe during the measurement sequence. Gross gamma-ray measurements were made by pushing the detector along in one (1) foot increments through a distance of eight (8) feet for each Core Flood line. The values recorded in Reference 22 were for both Core Flood lines. It was assumed that fuel material did not reach the CF-V5A valve or the straight run of piping that approaches

the valve from the reactor side. Some fuel material was detected at the CF-V5B valve and this quantity is included in the Reactor Building Miscellaneous PDSR. The quantity of fuel measured on the RV side of the CF-V5 valves was 0.6 kg for the CF-A line and 0.4 kg for the CF-B line which results in a total 1.0 kg UO<sub>2</sub>. The uncertainty was determined to be ±55% overall for the hot legs, cold legs, RC pumps and CF lines grouped together (Reference 12) and is reflected in Table 2.

#### 4.3 Cold Legs 1A/B and 2A/B

Radiation measurements were performed using small GM detectors inserted into the cold legs from the reactor vessel defueling work platform in accordance with Reference 23. The gross gamma-ray measurements were compared to the computer models generated by the Microshield computer code (Reference 14) to determine the estimated fuel quantities in each cold leg. The results were (Reference 12):

Cold leg 1A contains	3.3 kg UO <sub>2</sub>
Cold leg 1B contains	1.8 kg UO <sub>2</sub>
Cold leg 2A contains	3.9 kg UO <sub>2</sub>
Cold leg 2B contains	<u>2.4 kg UO<sub>2</sub></u>
Total	11.4 kg UO <sub>2</sub>

The computer model selected to represent the fuel debris was based on the video inspections performed during the gross gamma exposure rate measurements. In all cases a sediment region along the bottom of the horizontal pipe sections covering the lower 120 degrees of piping was selected for the computer model. The sediment was assigned a density and thickness, and represented the source of the radiation exposure for a one (1)-foot length of pipe at each

measurement location. Surface exposure rates from films and RCS water background were subtracted from the gross gamma measurement values.

In order to measure the reactor vessel for remaining fuel ( $UO_2$ ), it will be necessary to completely drain the reactor vessel while making passive neutron measurements. After the reactor vessel has been completely drained, it will likely be partially refilled using water from the OTSGs. The filling operation is expected to wash most of the residual fuel out of the '1B' cold leg and nozzle into the reactor vessel. Therefore, the fuel measured in the '1B' cold leg (1.8 kg) is subtracted from the cold leg total shown above. The new fuel estimate is 9.6 kg  $UO_2$ .

The uncertainty was determined to be  $\pm 55\%$  overall for the hot legs, cold legs, RC pumps and CF lines grouped together (Reference 12) and is reflected in Table 2.

#### 4.4 Hot Legs "A" and "B"

The hot leg sections of the RCS were measured in the same manner described for the cold leg sections in Section 4.3. The results were determined after the hot leg segments were defueled using a specially designed squeegee/vacuum tool to scrape and vacuum the debris in the hot legs. This tool covered the lower third of the hot leg diameter with the vacuum slot located on the bottom six (6) inches at the centerline, Reference 16. As with most of the RCS defueling/quantifying effort, the tooling was deployed from the defueling work platform on elevation 331' of the Reactor Building.

The results of the measurements taken after the defueling effort was completed indicate that the following fuel quantities remain in each hot leg (Reference 12):

"A" hot leg	0.9 kg UO <sub>2</sub>
"B" hot leg	<u>1.8 kg UO<sub>2</sub></u>
TOTAL	2.7 kg UO <sub>2</sub>

The uncertainty was determined to  $\pm 55\%$  overall for the hot legs, cold legs, RC pumps and CF lines grouped together (Reference 12) and is reflected in Table 2.

#### 4.5 Decay Heat Drop Line

The decay heat drop line was measured in the same manner described for the cold leg sections in Section 4.3. A model was developed using the Microshield computer code (Reference 14) that presented the source region as a one (1)-foot length of debris. A decay heat line sample was used to develop the isotopic loading and fuel concentration per unit weight of debris (Reference 20). The decay heat line net sample weight was reported to be 590 g and contained 46.6 g of UO<sub>2</sub> (8% UO<sub>2</sub> by weight). A three (3)-foot section of the decay heat horizontal line was measured, and the results were extrapolated over the remaining fifteen (15) feet of piping. The results indicate that the decay heat line contains 1.5 kg of UO<sub>2</sub>. The uncertainty for this value is considered to be the same as that determined for the previously reported RCS components (i.e.,  $\pm 55\%$ )(Reference 12).

The total fuel remaining in the decay heat drop line is:

1.5 kg UO<sub>2</sub>  $\pm 55\%$  uncertainty

#### 4.6 Pressurizer Lines

Extensive defueling operations were performed in the RCS with the goal of removing the majority of the fuel transported to the RCS as a result of the accident. These activities were generally successful. For example, defueling operations removed greater than 90% of the debris in the Pressurizer.

Debris in the Pressurizer Spray Line was flushed back into the Pressurizer and RCS Cold Leg 2A using water processed through the DWCS. Although the effort did not result in removing fuel from the primary system, it did relocate the fuel debris for ease of removal in subsequent defueling operations. Therefore, it was assumed that no measurable quantity of residual fuel remains in the Pressurizer Spray Line.

A portable directional survey instrument (Eberline Model HP-220A) was used to measure the exposure rate along the Pressurizer surge line. Higher intensities were noted along the eastward section of the surge line. The total quantity was derived for a uniform uranium deposition pattern over the east section of the surge line. The measured results determined a quantity of 0.2 kg UO<sub>2</sub> for the Pressurizer surge line (Reference 24).

The balance of lines connected to the Pressurizer were the safety and relief lines connected to the top of the Pressurizer tank. During the accident, relatively high velocity water was discharged through the PORV and safety valves to the RCDT. The velocity of the discharge flow was sufficient to keep micron sized fuel debris material in suspension and moving along the discharge pipe down to the RCDT. Inspections of the Pressurizer performed during defueling indicated there was only a small quantity of loose debris and that

most of the remaining fuel deposits consisted of large pieces of debris (approximately 5 cm wide x 10 cm long x 2.5 cm thick) (Reference 25). There was no indication of significant fuel deposits present in the drain line to the RCDT during the D-Ring surveillance type measurement programs.

The total fuel content for the Pressurizer lines was estimated to be 0.2 kg UO<sub>2</sub>. Since no uncertainty was reported for the results of this analysis, an uncertainty of ±60% was arbitrarily assigned to the fuel content value of the Pressurizer lines.

#### 4.7 Surface Films

Reactor fuel is present in the form of surface films on RCS components (Reference 9). Several RCS components were analyzed to determine the quantity of UO<sub>2</sub> present within these adherent and loosely adherent surface film layers. The manway covers from the Once Through Steam Generators (OTSGs) and the Pressurizer, as well as the 'A' OTSG hand hole cover were sampled. Surface materials were removed by scraping and analyzed for fuel content. Reference 9 presents the results of the scrape sample analyses and the total estimated fuel (UO<sub>2</sub>) present in the RCS as surface films. Under the assumption that the film characteristics of the three (3) manway plates are equally representative of the RCS, the films averaged 0.0019 inches thick and contained 18 µg/cm<sup>2</sup> of UO<sub>2</sub>. The total fuel (UO<sub>2</sub>) estimate was 4.6 kg. The assumed uncertainty for the surface film quantity was ± 60% (one sigma).



## 5.0 CONCLUSIONS

The estimate of record of the quantity of fuel ( $UO_2$ ) remaining in the Reactor Coolant System is 25.8 kilograms  $\pm$  43% (at one sigma). The uncertainty is based on a grouped uncertainty of  $\pm$ 55% for the hot legs, cold legs, RC pumps and CF lines and individual component uncertainties associated with the DH drop line, surface films and Pressurizer lines addressed in this Post Defueling Survey Report.

This estimate of record is derived from the measured data summarized in Table 2. The data shows that 61% of the remaining fuel is contained in two components, the Reactor Coolant Pumps and the Cold Legs. These components are expected to remain static once drain down is completed. No future water flows are planned except possibly to remove condensation or inleakage from the Reactor Building. Additional measurements of the RCS components are not warranted based on ALARA considerations due to the small quantity of fuel ( $UO_2$ ) measured to date. After final draindown of the RCS, the Reactor Building will be isolated from all systems outside the RB by maintaining containment isolation.

The goal of the TMI-2 defueling program was to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the 25.8 kg quantity of  $UO_2$  remaining in the RCS components is less than 4% of the total allowable fuel ( $UO_2$ ) remaining at TMI-2.

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19. GPUNC Engineering Calculation No. 4800-3212-89-010, Revision 0, Decay Heat Line Fuel Estimates, May 16, 1989.

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23. GPUNC Unit Work Instruction No. 4730-3255-89-R742, Cold Leg Characterization, August 1989.
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TABLE 1

ASSAY METHODS UTILIZED TO MEASURE RCS COMPONENTS IN THE REACTOR BUILDING

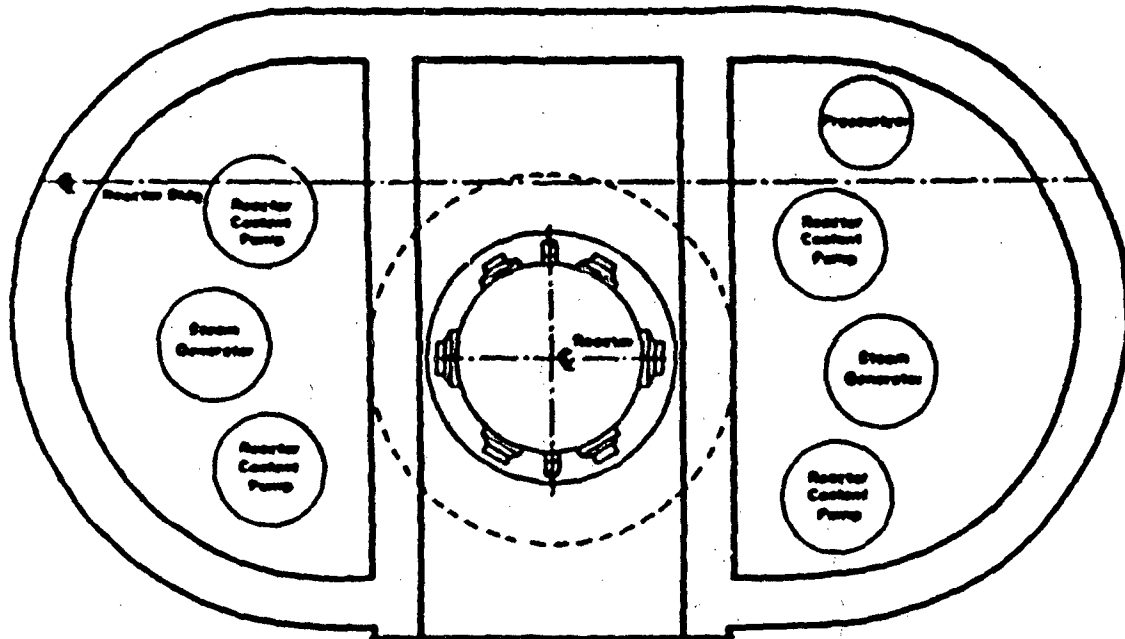
<u>Area Description</u>	<u>Assay Method</u>	<u>Calculation Number</u>
Reactor Coolant Pumps	Engineering Analysis	4249-3220-91-005
Core Flood Lines	Gross Gamma-Ray	4249-3220-91-005
Cold Legs 1 A/B and 2A/B	Gross Gamma-Ray	4249-3220-91-005
Hot Legs 'A' and 'B'	Gross Gamma-Ray	4249-3220-91-005
Decay Heat Drop Line	Gross Gamma-Ray	4800-3212-89-010
Pressurizer Lines	Gross Gamma-Ray	4550-3223-85-004
Surface Films	Beta, Alpha, Gamma, Sampled	Reference 9

TABLE 2

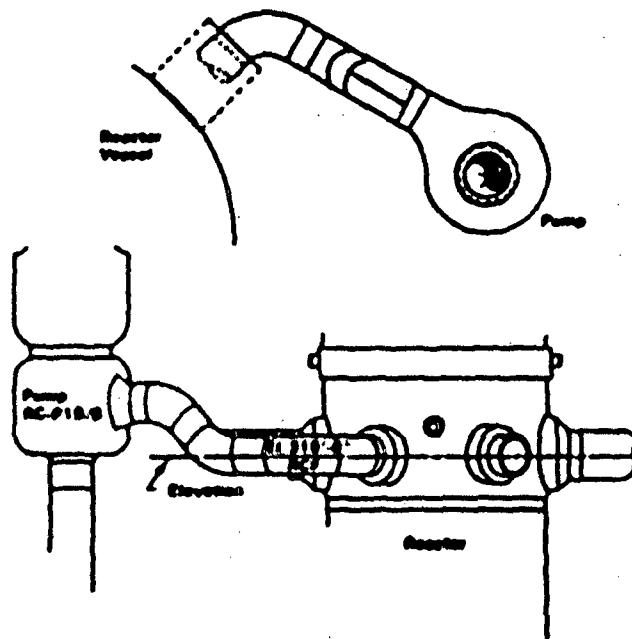
## SUMMARY OF SNM INVENTORY FOR RCS AREAS

<u>Area/Component</u>	<u>Fuel Quantity</u>	<u>Determined Uncertainty</u>	<u>Reference</u>
Reactor Coolant Pumps	6.2 kg	± 55%	PDSR Paragraph 4.1
Core Flood Lines 'A' and 'B'	1.0 kg		PDSR Paragraph 4.2
Cold Legs 1 A/B and 2 A/B	9.6 kg		PDSR Paragraph 4.3
Hot Legs 'A' and 'B'	2.7 kg		PDSR Paragraph 4.4
Decay Heat Drop Line	1.5 kg	± 55%	PDSR Paragraph 4.5
Pressurizer Lines	0.2 kg	± 60%	PDSR Paragraph 4.6
Surface Films	<u>4.6 kg</u>	<u>± 60%</u>	PDSR Paragraph 4.7
Estimate of Record and Uncertainty	25.8 kg	± 43%	

**FIGURE 1 - REACTOR COOLANT PUMPS**

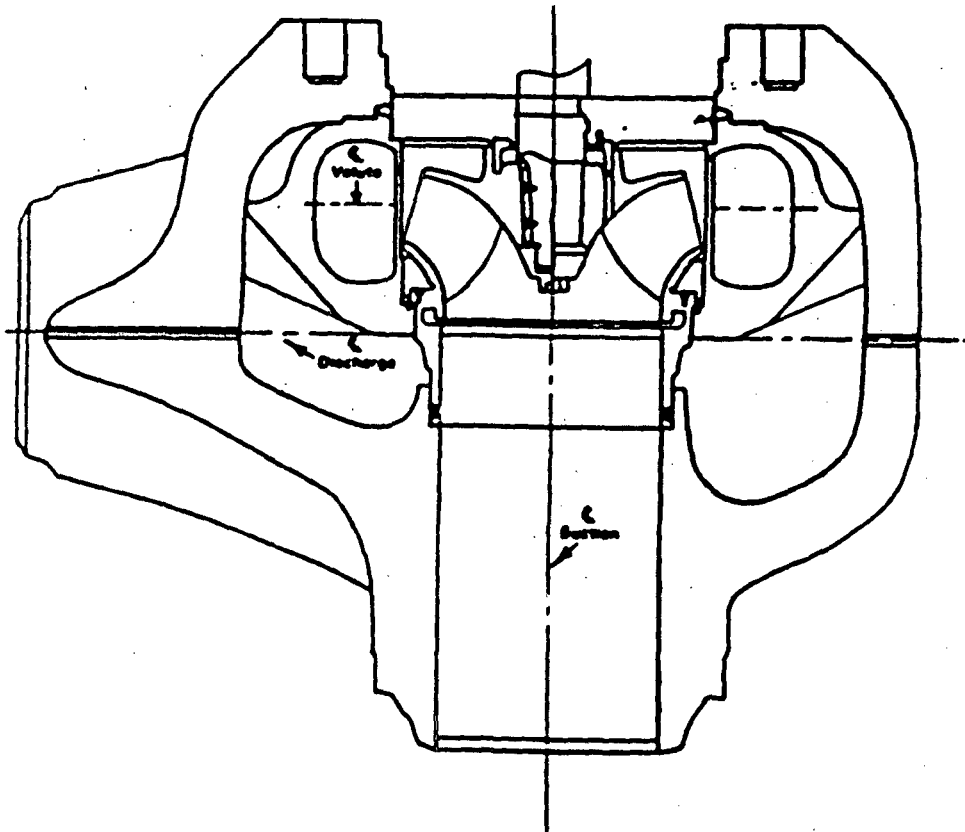


**REACTOR COOLANT PUMPS  
LOCATIONS INSIDE D RING**



**REACTOR COOLANT PUMP AND  
ADJACENT HORIZONTAL PIPING**

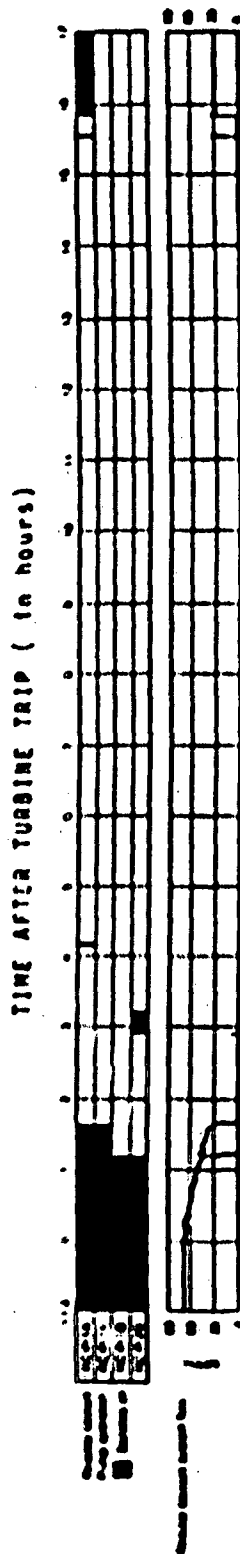
**FIGURE 2 - RC PUMP CROSS SECTION**



**REACTOR COOLANT PUMP CASING**



FIGURE 3 - RC PUMP TIME HISTORY



From Reference 2

FIGURE 4 - CORE FLOOD TANKS AND PIPING

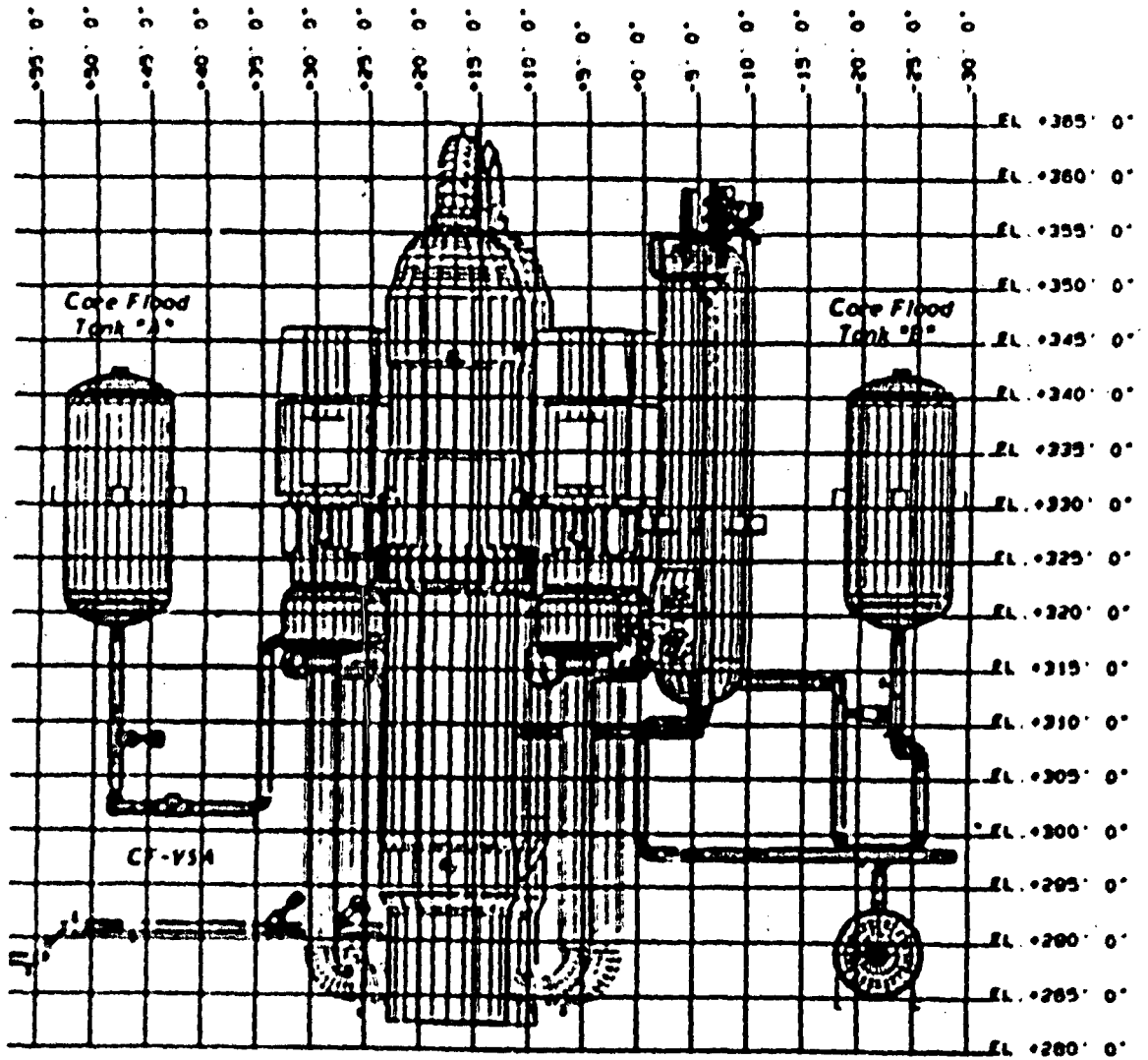


Diagram of RCS Showing Core Flood Tanks and Piping System

FIGURE 5 - CORE FLOOD PIPING

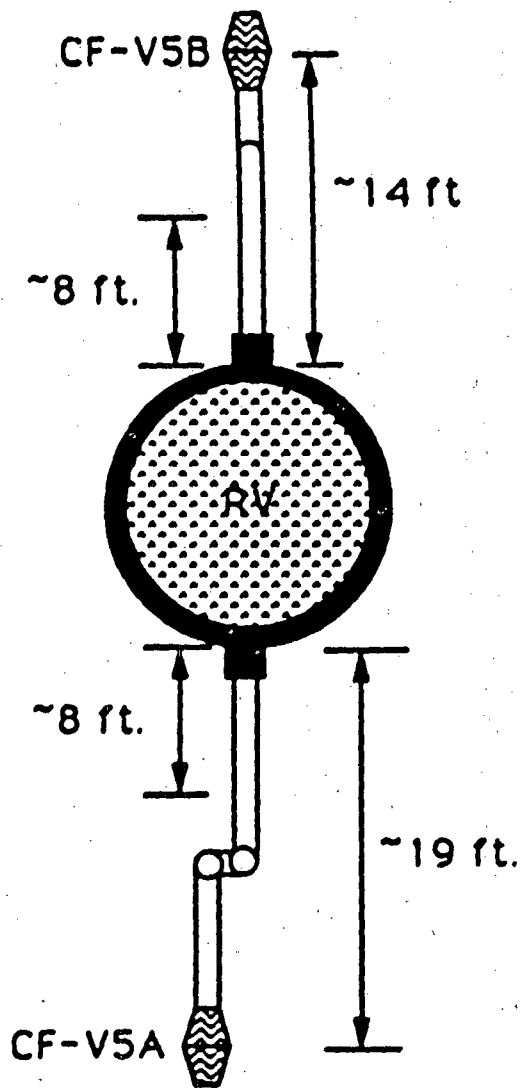


DIAGRAM SHOWING CORE FLOOD PIPING SYSTEM TO "5" VALVES

FIGURE 6 - RCS PIPING

**Reactor Coolant System Arrangement – Plan**

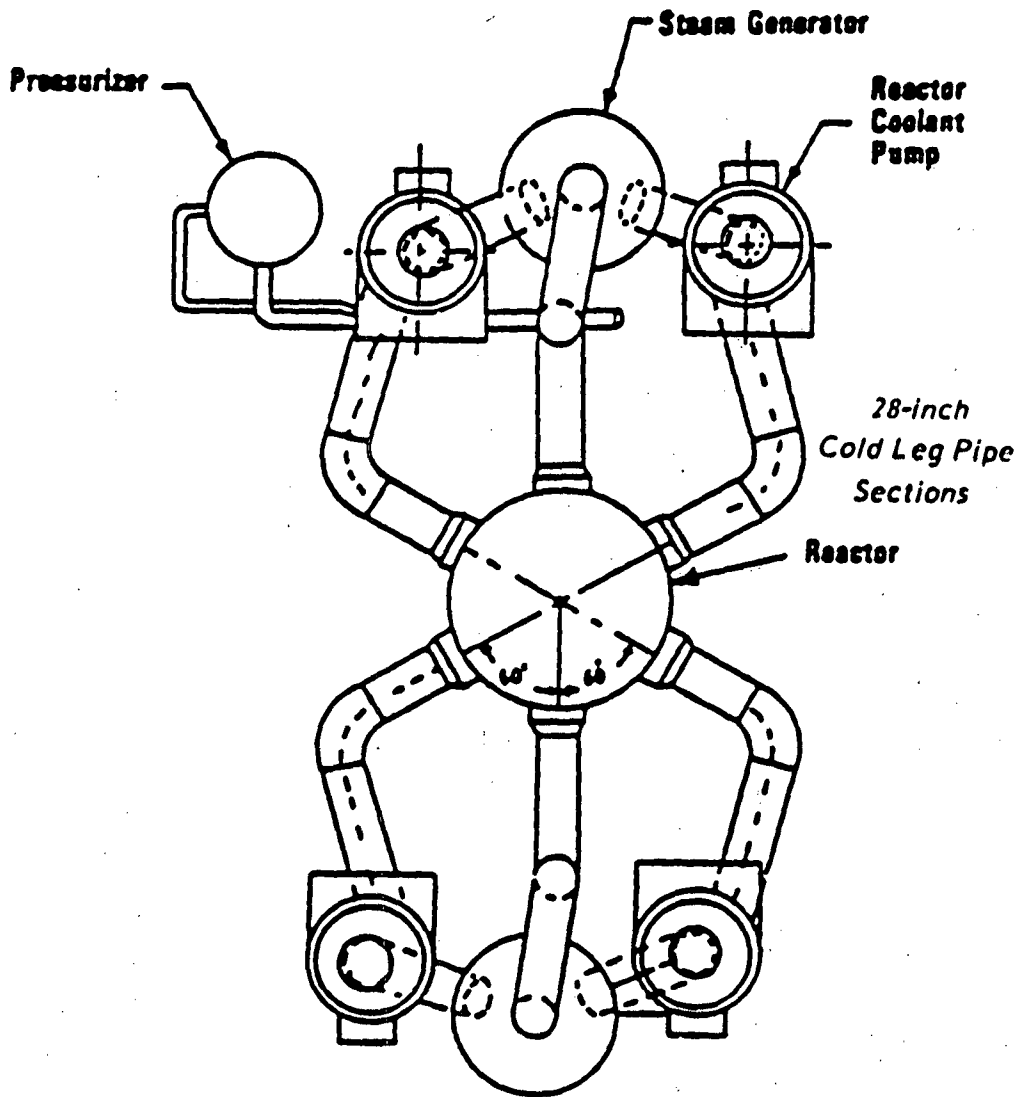
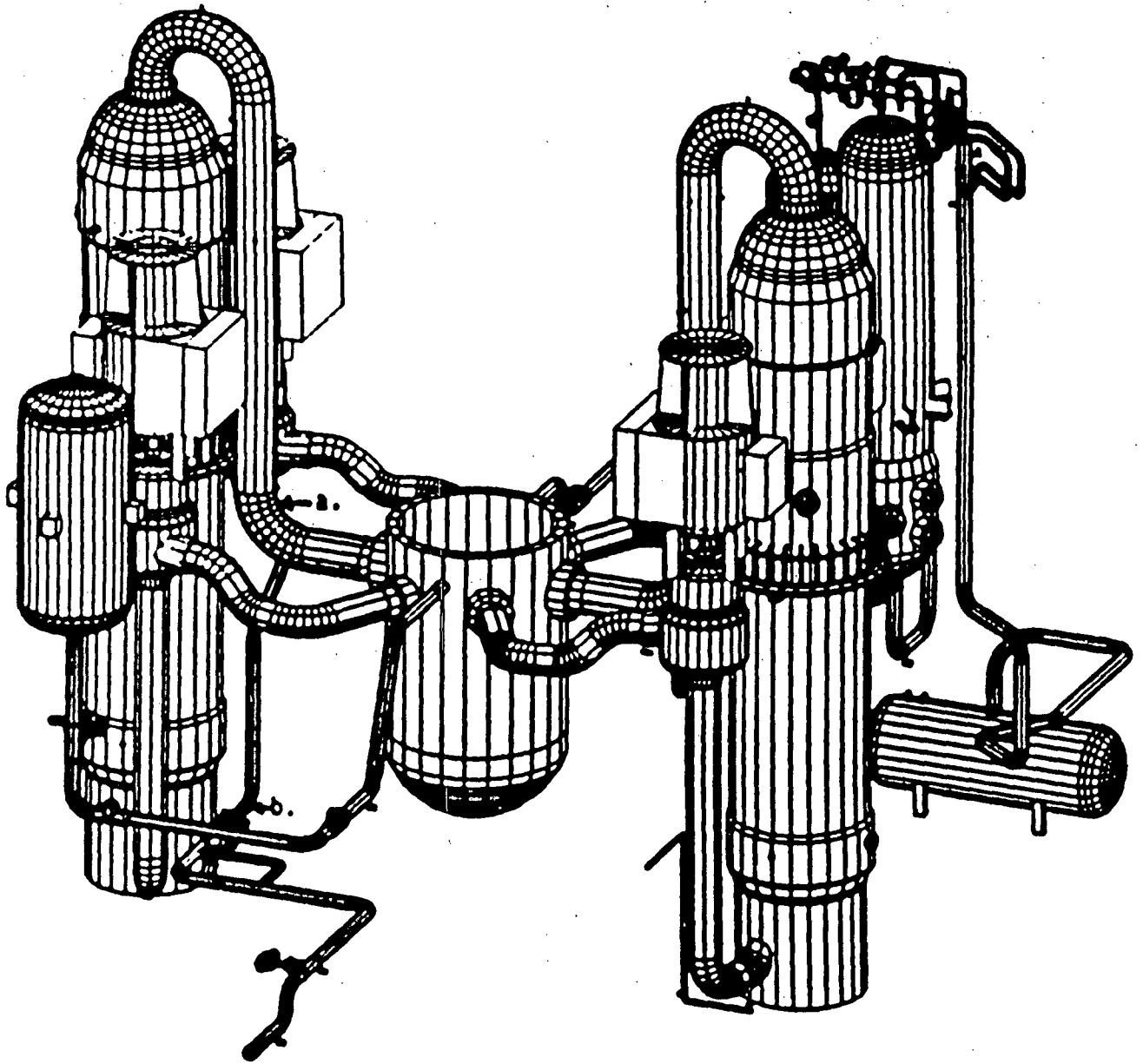
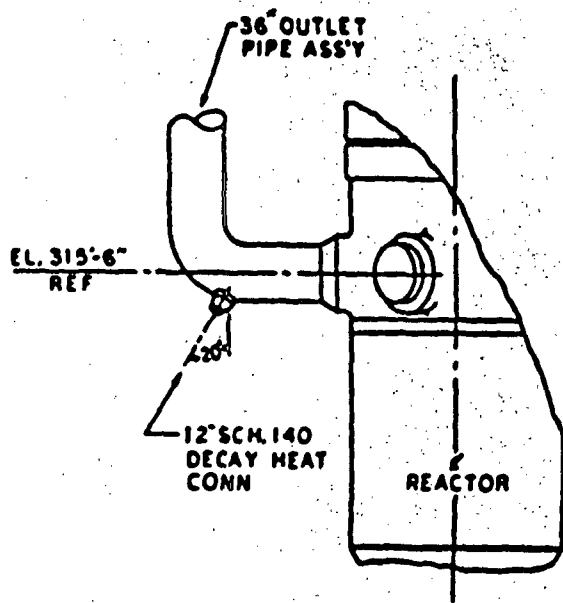


FIGURE 7 - RCS HOT LEGS AND DH LINE

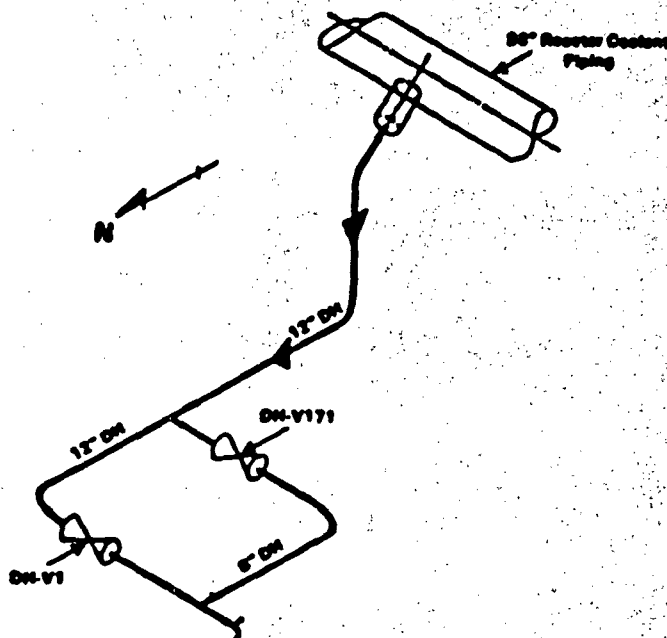


- a. "B" hot leg
- b. Decay heat line elbow

FIGURE 8 - DECAY HEAT DROP LINE



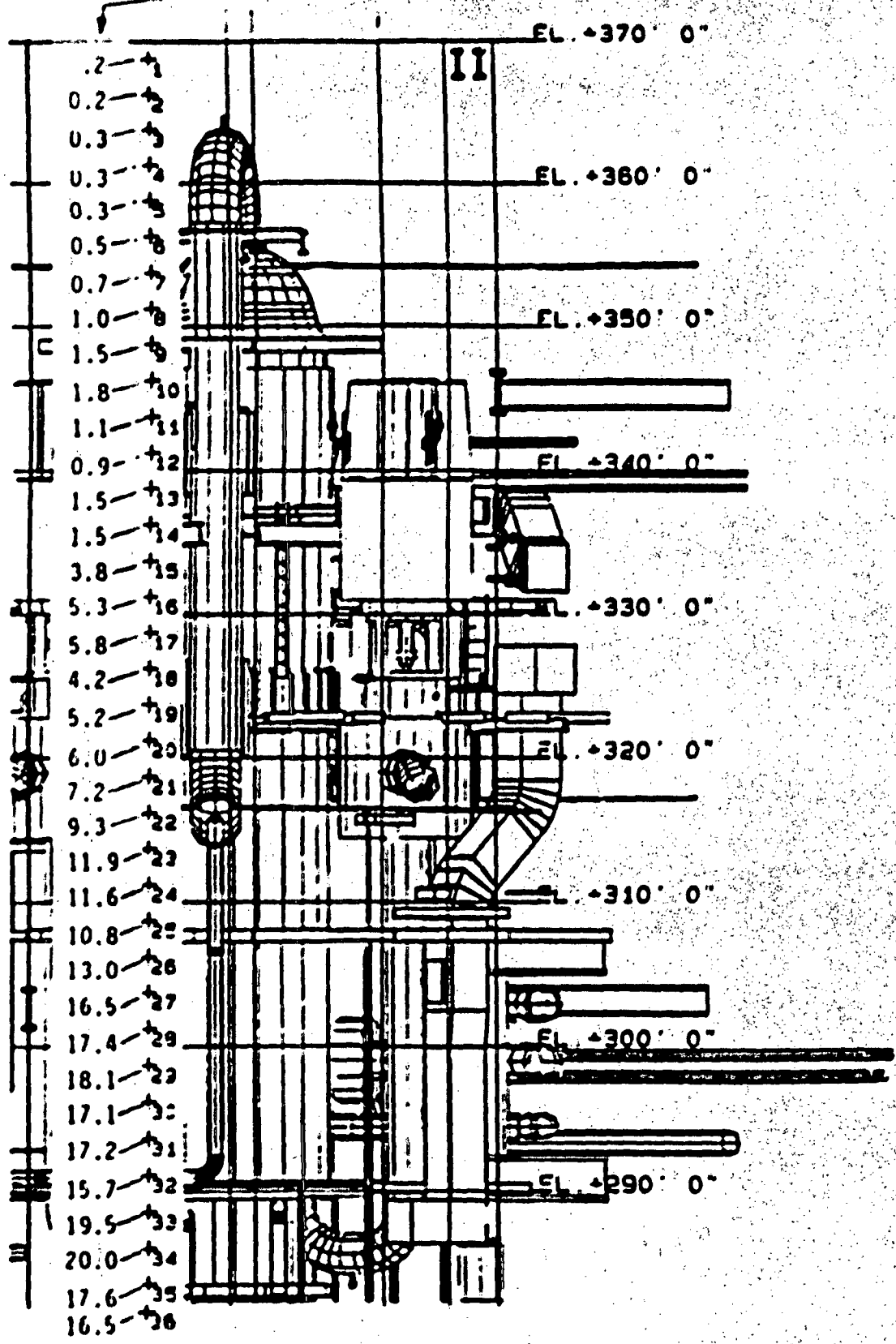
INTERSECTION OF DECAY HEAT NOZZLE WITH REACTOR COOLANT OUTLET




DECAY HEAT REMOVAL SYSTEM

FIGURE 9 - EXPOSURE RATES IN 'B' D-RING

TLD STRING (R/HR)



Buckley, John | Friday, February 10, 2012

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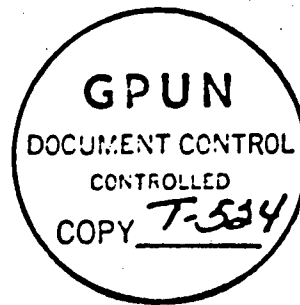
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SNM Accountability PlanRevision No.  
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TMI-2 SNM Accountability ProgramResponsible Office  
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E1-4	0-00						
E1-5	0-00						
E1-6	0-00						
E1-7	0-00						
E1-8	0-00						



	Signature	Concurring Organization Element	Date
Originator	<i>John Schuch</i>	Mgr., SNM Accountability	3/2/87
Concurred by	<i>R. Warren</i>	RTR	3/17/87
	<i>R. S. Daniels / SI</i>	Mgr., Project Planning and Analysis	3/25/87
	<i>J. J. Bencic</i>	SRG	3/27/87
	<i>E. E. Munn</i>	Supv., Document/Modification Control	3-27-87
	<i>W. Marshall / W. Miller</i>	Mgr., Plant Operations	3/27/87
	<i>J. J. Bencic</i>	Mgr., Recovery Engineering	3-27-87
	<i>J. J. Bencic</i>	Radiological Controls Director	3-27-87
	<i>J. J. Bencic</i>	Mgr., Waste Management	3-27-87
	<i>J. J. Bencic</i>	Mgr., Licensing	3/27/87
	<i>J. J. Bencic</i>	Mgr., TMI QA Mod/Ops	3-27-87
Approved by	<i>J. J. Bencic</i>	Director, Licensing and Nuclear Safety	3-27-87
	<i>Thomas J. Bennett</i>	Site Operations Director	3/30/87
		OOD	4/2/87

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<b>GPU Nuclear</b>	<b>TMI-2 Unit Policy/Plan</b>	Number <b>4000-PLN-4420.02</b>
Title <b>SNM Accountability Plan</b>		Revision No. <b>0-00</b>
<b>1.0 PURPOSE</b>		
<p>1.1 The purpose of this plan is to describe the Three Mile Island Unit 2 (TMI-2) Special Nuclear Material (SNM*) Accountability Program. This plan identifies the methods and sequence of SNM accountability, the Quality Assurance Program that will be built into SNM measurement activities, the areas, systems and components that will undergo formal SNM measurement and the areas, systems and components that do not require SNM assessment. The plan identifies the TMI-2 organizations that will directly perform SNM assessment and the organizations that will provide significant support. The plan also describes how programmatic ALARA will be implemented in SNM assessment activities.</p>		
<p>*For the purposes of this plan, the term SNM will be utilized to describe the residual fissile material derived from the original enriched UO<sub>2</sub> fuel.</p>		
<p>1.2 The post-defueling survey of the TMI-2 plant for residual special nuclear material will be performed by the implementation of this plan. As identified in this plan, the post-defueling survey is a process by which the entire TMI-2 plant will be reviewed to identify areas known to contain special nuclear material or that could contain SNM and the presence and quantity of special nuclear material (SNM) in each area will be determined. The accomplishment of the SNM measurements and associated engineering analysis will constitute completion of the post-defueling survey. This plan, the SNM Accountability Plan, describes the process by which the post-defueling survey will be conducted.</p>		
<b>2.0 BACKGROUND</b>		
<p>2.1 The March, 1979 accident resulted in significant damage to the core and in subsequent release of fuel and fission products into the Reactor Coolant System and other closely related systems. The TMI-2 core currently consists of loose fuel pellets, solidified fuel, structural metal components (e.g., end fittings), loose rubble and partial fuel assemblies. This collection of material is generically referred to as core debris. As a result of the core condition, fuel accountability by the normal method of counting individual fuel assemblies is not possible.</p>		
<p>2.2 Core debris is presently being loaded in special canisters and shipped to the Department of Energy Idaho National Engineering Laboratory (DOE INEL) facility in Idaho. Each shipment is accompanied by a Nuclear Material Transaction Report (DOE/NRC Form 741) which shows the net weight of the contents of each canister and a best available physical description of the contents. A statement that quantification of the amount of SNM in each canister is not possible also accompanies each shipment as an annotation on the DOE/NRC Nuclear Material Transaction Report Form 741.</p>		

<b>GPU Nuclear</b>	TMI-2 Unit Policy/Plan	Number 4000-PLN-4420.02
Title SNM Accountability Plan		Revision No. 0-00
<p>2.3 The canister contents are a mixture of SNM and other core debris. There is no feasible method at TMI-2 to determine the exact content of fuel in each canister. Therefore, SNM accountability for TMI-2 will be based on the total measured SNM remaining in the plant after defueling is complete. A final plant inventory of residual SNM will be reported on the DOE/NRC Material Balance Report (DOE/NRC Form 742).</p> <p>2.4 In October, 1985, GPU Nuclear, the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) entered into an agreement (Reference 11.1 and 11.2) that final SNM accountability for TMI-2 would be performed after defueling was completed and would be based upon a thorough post-defueling survey of TMI-2. This post-defueling survey would quantify, as accurately as possible, the amount of residual SNM in plant systems and components. Implied in this agreement was the understanding that the post-defueling survey would involve all areas, structures, systems and components where SNM could possibly have been deposited as a result of the 1979 accident and subsequent recovery activities.</p> <p>3.0 <u>SUMMARY</u></p> <p>3.1 Formal SNM assessment activities are currently scheduled to begin in 1987. A measurement schedule is presented in Appendix 1. In summary, the SNM assessment schedule is based upon the completion of defueling activities in the Reactor Building components and gross decontamination of the selected Auxilliary and Fuel Handling Building systems and associated cubicles. The current schedule calls for SNM accountability to be completed after Reactor Coolant System (RCS) draindown has occurred. The projected SNM accountability schedule is based on current defueling and decontamination schedules. The schedule will be adjusted as needed to reflect cleanup program progress.</p> <p>3.2 SNM measurements will be performed as areas, systems and components are placed into an isolated configuration that ensures no fuel transport in or out after the SNM survey has been completed. The configuration will be selected to enhance SNM detection with due regard for system bounds, piping configuration and measurement requirements. Following SNM survey, the configuration will be administratively and physically controlled. If the configuration is modified in a manner that could result in SNM transport, suitable measurements will be performed to ensure accurate accountability.</p> <p>3.3 The entire TMI-2 plant has been reviewed to determine where SNM may have been deposited as a result of the 1979 accident and subsequent recovery activities. Locations have been placed into three categories: Category 1 - locations where SNM is definitely deposited; Category 2 - locations where it can be reasonably postulated that SNM may be deposited; and Category 3 - locations where it can be shown that SNM was not deposited. Appendix 1 also identifies each area classification.</p>		

3.4 All Category 1 areas will undergo SNM survey. Category 2 areas will undergo SNM survey after surface decontamination and/or system flush activities are completed. Category 3 areas will be identified as not requiring SNM assessment based upon authoritative analyses of the TMI-2 accident (NSAC 80-1: Analysis of Three Mile Island - Unit 2 Accident; Rogovin Report: Three Mile Island, A Report to the Commissioners and the Public) and a review of recovery activities.

NOTE: Some areas of the plant may be reclassified as a result of ongoing or future recovery activities.

3.5 SNM accountability at TMI-2 will be a complex task. Inaccessibility of some systems and components, high area radiation backgrounds, complex geometries and the required indirect measurement of fuel will complicate physical measurement of SNM quantities. Also, selected TMI-2 systems cannot be surveyed until RCS draindown occurs. Therefore, several alternative techniques for performing measurement of SNM quantities will be useful. Reference 11.3 describes those techniques. Appendix 1, as noted above, classifies plant areas and, where possible, specifies the method(s) to be used to assess each listed area, system or component.

3.6 SNM assessment is an Important to Safety (ITS) activity. The TMI-2 Recovery QA Plan applies to SNM assessment activities. QA/QC will review and approve the SNM accountability plan and SNM measurements procedures and Unit Work Instructions. Measurement equipment will be maintained and calibrated in accordance with Quality Assurance/Quality Control (QA/QC) requirements. Individual SNM assessment activities will include QA/QC verification of essential parameters as deemed necessary. Records of SNM assessment activities and associated analyses will be subjected to QA/QC monitoring and auditing. Engineering calculations for SNM assessment will be performed in accordance with the TMI-2 Engineering Calculation Procedure 4000-ENG-7310.02 and will be independently audited.

4.0 SNM ACCOUNTABILITY PROCESS

4.1 The SNM Accountability Process will establish the quantity of residual SNM at TMI-2 after defueling is completed. The quantity of residual SNM will be determined through measurements, sampling and engineering analysis. The determination of the quantity of SNM in a specific area, system or component will be documented in a SNM assessment package. The SNM assessment package, will contain, as a minimum: a detailed description of the area, system or component; its role in the accident and/or recovery activities; the rationale supporting a conclusion as to whether the possibility of contained residual SNM exists; and if so, a SNM measurement document or previous fuel characterization results and an appropriate engineering calculation. The purpose of each set of data contained in the SNM assessment package is detailed below:

4.1.1 Description - The purpose of this section will be to provide detailed information on the area, system or component being assayed. Included will be a description of each significant piping section, component and surface; a description of the

accident and recovery program history of the above; and available photographs and/or drawings. Finally, if an area, system or component is determined to have no possibility for containing residual fuel, it will be so stated, with supporting rationale, in the description section.

- 4.1.2 SNM Measurement - Specific measurements of residual SNM quantities will be performed utilizing either QA approved procedures or Unit Work Instructions. All SNM measurement documents will contain data sheets which will record the measurement data, equipment calibration information and essential supporting information. All SNM measurements will be performed in accordance with appropriate industrial safety requirements.
- 4.1.3 Fuel Characterization Measurements for SNM Accountability - Several plant areas and components were surveyed for residual SNM deposits prior to initiation of the formal SNM accountability program. In some cases, it will be advantageous to utilize the results of these previous measurements; personnel radiation exposure will be minimized. Previous fuel characterization measurements will be utilized for SNM accountability purposes at the discretion of the Manager, Post Defueling Survey and SNM Accountability under the following conditions:
- a. The area, system or component measured has been maintained in a fixed configuration, from the time of the measurement to the present, that precludes the possibility of transport of SNM into or out of the area, system or component.
  - b. The area, system or component can continue throughout the duration of the SNM accountability measurement program to be maintained in a configuration that precludes the transport of SNM into or out of the area, system or component and appropriate controls are in place to ensure configuration controls.
  - c. The previous fuel characterization SNM measurement package is accepted by Quality Assurance/Quality Control.
- 4.1.4 SNM Assessment Engineering Calculation - The SNM assessment engineering calculation will be the documented engineering analysis which determines the quantity of residual SNM in an area, system or component based on the raw measurement data. SNM engineering calculations will be performed in accordance with Procedure 4000-ENG-7310.02, Engineering Calculations. The calculation will determine the quantity of SNM in a given location based upon the configuration of the object assayed, the analysis of the survey data and the measurement instrumentation performance capabilities. Also relevant to the determination of

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the quantity of residual SNM will be the specific assumptions deriving from a review of the measurements made and an analysis of the accident history and recovery efforts relating to the area, system or component.

#### 4.2 Determination of SNM in Radioactive Waste and Sample Shipments

4.2.1 The total amount of SNM shipped off-site as radioactive waste and/or as samples will be determined in accordance with approved Important to Safety (ITS) procedures. The amount of SNM shipped will be recorded on the appropriate shipping forms and the NRC Form 741 Nuclear Material Transaction Report. The quantities of SNM reported as shipped will be summed for input into the final SNM accountability process.

#### 4.3 Final SNM Accountability

Final SNM accountability will be performed by summing the residual SNM quantities identified in the individual SNM measurements and reporting the sum quantity as the remaining plant inventory of special nuclear material. The amount of fuel shipped to the Department of Energy (DOE) Idaho National Engineering Laboratory (INEL) will be determined by subtracting the sum of the final plant inventory and the amount of SNM shipped as radioactive waste from the total plant inventory of SNM as reported on the most recent SNM Material Balance Report (NRC/DOE Form 742) as corrected for decay.

- Last Reported Inventory
- Decay correction
  - Final In-plant inventory
  - SNM shipped as samples and Radwaste
  - SNM shipped to INEL in fuel, filter and knockout canisters

#### 5.0 SNM MEASUREMENTS

- 5.1 SNM measurement will be performed on TMI-2 Category 1 and Category 2 structures, systems and components (Appendix 1). A SNM measurement will be performed on each individual location once it is placed in a configuration for Post-Defueling Monitored Storage. A determination of the residual SNM in each location will be based on individual SNM measurements performed using approved procedures or by examination and analysis of previously performed fuel characterization measurements.
- 5.2 SNM measurements will be performed in accordance with an approved procedure or Unit Work Instruction when existing data from previous fuel characterization measurements are insufficient for final SNM accountability. Individual SNM measurements will be performed in accordance with generic measurement procedures or a specific Unit Work Instruction. Generic procedures will be utilized for measurements performed utilizing a



standard technique (e.g., Gamma Spectroscopy). Unit Work Instructions will be utilized for measurements that require special, one of a kind, techniques (e.g., OTSG tube film SNM measurement). Data sheets attached to each SNM measurement document will be used to record the data required for post measurement analysis.

- 5.3 SNM measurement documents shall be reviewed and approved in accordance with TMI-2 Unit Procedure 4000-ADM-1218.02, TMI-2 Document Evaluation, Review and Approval. SNM measurement documents will require concurrence by the following organizations, as a minimum: Plant Operations, Quality Assurance/Quality Control, SRG and SNM Accountability. SNM measurement Unit Work Instructions will require the additional concurrence of Radiological Controls. In addition, each SNM measurement document will be reviewed by all organizations from whom support is required. The Site Operations Director shall be the approver of SNM Measurement documents.
- 5.4 SNM measurement documents shall be archived in CARIRS. Copies of all data sheets will be submitted to Data Management and Analysis for post measurement analysis.

#### 6.0 SNM MEASUREMENT TECHNIQUES

- 6.1 As stated in the introduction, the post-defueling SNM assessment at TMI-2 will be a complex task. Several different measurement techniques will be used. Technique selection for each measurement will depend upon the configuration of the component or area to assayed, physical access limitations, area radiation dose rates and the likely nature of the form of special nuclear material (fuel) deposits. Current plans are to make extensive use of gamma scintillation counting, visual inspections, scrape sampling of films deposited on metal surfaces and gas proportional detection of alpha radiation. Several areas may be surveyed utilizing two or more techniques. A detailed description of the measurement techniques and selection criteria can be found in Reference 11.3.

#### 7.0 QUALITY ASSURANCE FOR SNM ACCOUNTABILITY

- 7.1 The results of the SNM Accountability Program will be the basis for final SNM accountability at TMI-2. In addition, final SNM accountability may be a highly visible element of the completion of the TMI-2 Cleanup Program. Therefore, the SNM accountability activities are classified as "Important to Safety" and shall comply with the TMI-2 Recovery QA Plan.
- 7.2 SNM accountability will be based on a determination of the quantity of residual Special Nuclear Material (SNM) remaining in the TMI-2 areas, systems and components after defueling and water processing activities have been completed. The determination of the residual SNM quantities will be based upon measurements performed utilizing QA/QC approved procedures or upon measurement packages that contain previously performed fuel characterization measurements reviewed and approved by QA/QC on an after-the-fact basis.

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- 7.3 SNM measurement activities performed via approved procedures or Unit Work Instructions will be reviewed by QA/QC for inclusion of hold/witness points. Specific activities (e.g., critical items) may also be identified by the document preparer which require QA/QC verification. Activities requiring QA/QC observation will require adequate notification to ensure that QA/QC support is available for the specific activity.
- 7.4 SNM engineering calculations will be performed in accordance with procedure 4000-ENG-7310.02, Engineering Calculations. Data utilized in the engineering calculations will be obtained from the completed QA approved procedure or Unit Work Instructions data sheets or from QA reviewed and approved data acquisition measurement packages. The calculations will be independently verified in accordance with the procedure. SNM measurement packages that identify residual SNM deposits greater than 1 kilogram will be submitted for a separate independent review to an organization other than Data Management and Analysis.
- 7.5 Computer codes utilized to quantify residual special nuclear material will be verified by Data Management and Analysis by benchmarking with accepted industry codes. Verification will be documented. Only verified and approved computer codes will be utilized. Approved code versions will be controlled to preclude unauthorized modification. Code versions utilized in engineering calculations will be specifically identified.
- 7.6 Equipment utilized to quantify residual special nuclear material via QA approved procedure or Unit Work Instruction will be calibrated and operated in accordance with these procedures. Essential equipment identification information (e.g., type, size, configuration) and performance data (e.g., counts, duration of count, location of detector) will be recorded on the data sheets.
- 8.0 CONFIGURATION CONTROL OF AREAS, SYSTEMS OR COMPONENTS CONTAINING RESIDUAL SNM**
- 8.1 In order to ensure that the SNM measurement process is accurate, controls must be established to ensure that special nuclear material (SNM) is not "double counted". Double counting could occur when SNM is relocated out of a component that has already been measured into a radioactive waste shipment or a component still requiring SNM assessment.
- 8.2 Administrative controls will be utilized to maintain physical isolation of areas, systems or components that have undergone SNM measurement so that transport of SNM into or out of the area, system or component is precluded. The type of administrative controls will depend upon the nature of the component.
- 8.3 Piping systems and components connected to piping systems (e.g., pumps, tanks) will be isolated utilizing "red" tags via the TMI-2 Switching and Tagging Procedure 4000-ADM-3020.04. Red tags prohibit the changing of position of a component (e.g., valves, electrical breakers). "Red tagged" components will be isolation barriers intended to prevent the transport of

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residual SNM into or out of areas, systems or components. If an isolation barrier is removed (i.e., a valve opened or other violation of configuration control), the need for repeat measurement of SNM in the affected piping will be evaluated.

- 8.4 Open areas (e.g., Reactor Building [RB] basement floors, Auxiliary Building sump) will be controlled by one of two methods to prevent fuel transport. Areas will be maintained in a dry condition so that liquid cannot displace residual SNM or, if subjected to liquid flow, liquid effluents will be sampled. Sampling of the liquid effluent for the presence of residual SNM will be performed in accordance with QA approved procedures.

#### 9.0 ALARA FOR SNM ACCOUNTABILITY

- 9.1 The program for SNM accountability will be conducted in accordance with the "As Low As Reasonably Achievable" (ALARA) principle for radiation exposure. The SNM accountability program will result in the exposure of personnel to ionizing radiation when SNM measurements are performed in the Auxiliary, Fuel Handling or Reactor Building. Personnel radiation exposures will be maintained ALARA by limiting the number of measurements to those essential for SNM accountability and by planning each measurement to minimize personal exposure.
- 9.2 The number of required SNM measurements will be limited by utilizing, when possible, previously performed fuel characterization measurements. In addition, SNM measurements will be limited to those areas, systems or components which conceivably contain fuel.
- 9.3 The radiation exposure received by personnel performing SNM measurements will be kept ALARA by proper planning. Individual measurements will be designed to include efficient use of time in radiation areas, incorporate lessons learned on dose minimization from previous measurements and include the use of remote equipment when possible. Where possible, SNM measurements will be coordinated with radiological end point verification surveys in high radiation areas.

#### 10.0 RESPONSIBILITIES

- 10.1 The SNM Accountability program will be directed and controlled by the Licensing and Nuclear Safety Department. Specifically, the Manager, Post-Defueling Survey and SNM Accountability is responsible to develop and implement the SNM Accountability Plan which will control the post-defueling survey of TMI-2.
- 10.2 The Data Management and Analysis Section of the Project Planning and Analysis Department will develop procedures and techniques for performing individual SNM measurements, perform SNM measurements and, based upon data obtained, determine residual SNM quantities through formal engineering calculations.

- 10.3 The Licensing Section will provide major support to the SNM Accountability program. Licensing will develop the appropriate strategy for compliance with NRC and DOE regulatory requirements and will be the primary interface with the NRC in the review of the SNM accountability documents. Finally, Licensing will support submission of the final SNM accountability results and negotiate resolution of the final accountability/transfer of accountability of the TMI-2 Core to the DOE.
- 10.4 The Defueling Support Section will provide the major in-plant labor support for SNM accountability activities. Defueling Support will also provide scheduling and other administrative support.
- 10.5 The Radiological Controls Department will provide support to the SNM accountability program to ensure activities are conducted in a manner consistent with GPU Nuclear ALARA objectives.
- 10.6 Site Operations will provide support to SNM accountability by establishing and maintaining configuration control of the plant systems. The Site Operations Director shall be the approver of all SNM assessment procedures and Unit Work Instructions. In addition, the Radwaste Section of Site Operations will provide support for SNM measurements in the Auxiliary and Fuel Handling Buildings.
- 10.7 QA/QC will review SNM assessment documents and perform QA/QC inspection of SNM assessment activities.

## 11.0 REFERENCES

- 11.1 DOE letter of October 8, 1985 from W. W. Bixby to H. M. Burton (EG&G). "Accountability for the TMI-2 Core - HWB-100-85"
- 11.2 NRC letter of October 17, 1985 from B. J. Snyder to F. R. Standerfer (GPUNC). "Approval of Exemption from 10CFR30.51, 40.61, 70.51(d), and 70.53"
- 11.3 Technical Planning Department. January 1987. Instrument Selection for Residual Fuel Measurements. TPO/TMI-187, Revision 0. Middletown, PA; GPU Nuclear Corporation
- 11.4 Technical Planning Department. August 1985. TMI-2 Core Accountability. TPO/TMI-035, Revision 1. Middletown, PA; GPU Nuclear Corporation

## 12.0 Appendix 1

APPENDIX 1 - Area Classification and SNM Assessment Schedule

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
AXB01	BB Emerg. Boos. Pump	3	2uR7					
AXB02	Access Corridor (Drains)	3	2uR7					
AXB03	Access Area (Drains)	3	2uR7					
AXB04	Seal Injection Valve Room	2	25R7 (LMRA)	NaI Detector	3/87		5/87	Gamma Spectr. scheduled at later date
AXB05	H.U. Pump - 1C	1	20uR7 (LMRA)	NaGe Detector	3/87	1/87	2/87	Pre-flush status: 0.41 grams fuel* (TB-05-33)
AXB06	H.U. Pump - 1B	1	100uR7, 600uRS- (LMRA)	NaGe Detector	4/87	1/87	2/87	Pre-flush status: 23.4 grams fuel* (TB-05-33)
AXB07	H.U. Pump - 1A	1	50uR7, 100uRS- (LMRA)	NaGe Detector	5/87	1/87	2/87	Pre-flush status: 10.0 grams fuel* (TB-05-33)
AXB08	Spent Resin Storage Tank - 1B	2	0uR7	NaGe Detector				
AXB09	Spent Resin Storage Tank - 1A	2	0uR7	NaGe Detector				
AXB10	Spent Resin Storage Tank Pump	2	0uR7	Documentation				
AXB11	Aux. Sump Pump Valve Room	1	20uR7	Documentation			***	Pre-flush status: Gamma Spectr. shows 1.5 grams of fuel*. (TB-06-28; MDL-44A, 4B)
AXB12	Aux. Bldg. Sump Tank Room	1	700uR7, 40uRS- (LMRA)	NaI or NaGe Detector	4/87	5/87	6/87	Gamma Spectr. shows 262 grams* to sump, 1.4 grams* in sump tank and 1.9 grams* in pump (TB-06-28)
AXB13	Evap. Cond. Tanks, Pumps	3	4uR7					

NOTES: \*Best estimate of SNM quantity; measurement uncertainty described in the referenced Technical Bulletin or GEND document.  
 \*\*Area designations defined in Technical Data Book, TPO/TMI-009.  
 \*\*\*Will utilize existing documentation.

SNM CATEGORIES: 1. Known Fuel Present  
 2. Possible Fuel Present  
 3. No Fuel Present

LEGEND: LMRA - Labeled High Radiation Area  
 TBD - To Be Determined

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APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
A2D14	RC Evap. Room	3	100mR (LMRA)					
A2D15a	Cleanup Filters Room	3	10mR					NEL-F44B and NEL-P24B (c4.7 grams per filter) (TB-86-48)
A2D15b	Cleanup Filter After Room	3	10mR					
A2D16	Cleanup Dumb. - 2A	3	5R (LMRA)					
A2D17	Cleanup Dumb. - 2B	3	50R (LMRA)					
A2D18	Waste Transfer Pump Room	3	20mR, 100mR-					
A2D19	Waste Disposal Liquid Valves	1	5mR	NaI Detector or Documentation				
A2D20	RC Bleed Tanks 1B, 1C	1	5mR, 10mR- (LMRA)	NaI Detector	5/87	12/87	1/88	Possible small amounts of fuel (MCL-71B)
A2D21	RC Bleed Tank 1A	1	12mR	NaI Detector			1/88	Tank cleaned - presently contains no fuel
A2D22	North Stairwell	3	1mR					
A2D23	Elevator Shaft	3	20mR, 50mR-					
A2D24	Aux. Bldg. Sump Filters	2	60mR	NaI Detector or Documentation	11/87	10/87	1/88	Filter cartridges removed (no fuel in filters)
A2D26	Seal Injection Filters (MU-F4A, 4B)	2	60mR					Needs to be surveyed (filter cartridges removed, no fuel in filters)
A2D27	South Stairwell	3	2.5mR					
A2I01	Radwaste Disposal Panel	3	2mR					
A2I02	WB Sump Pump Filters	2	300mR	TBD		5/87	7/87	Not surveyed
A2I03	MCC 2-11EB	3	<2.5mR					

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**APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)**

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Bacon Date	System Flush Date	SNM Measurement Date	Remarks
AX104	MCC 2-21EB	3	< 2.5mR/h					
AX105	Substation 2-11E	3	< 2.5mR/h					
AX106	Substation 2-21E	3	< 2.5mR/h					
AX107	MCC 2-11EA	3	< 2.5mR/h					
AX108	MCC 2-21EA	3	< 2.5mR/h					
AX109	Mec. Services Coolers and Pumps	3	< 2.5mR/h					
AX110	Intermediate Coolers	3	< 2.5mR/h					
AX111	Intermediate Cooling Pumps and Filters	3	< 60mR/h					
AX112	Seal Return Coolers and Filters (MU-F-3)	2	300mR/h (LMRA)	NaI or NaCo or Documentation	4/87	1/87	8/87	Needs to be surveyed (Filter Cartridge removed)
AX113	Waste Gas Analyzer	3	<100mR/h (LMRA)					
AX114	MUBP Dmain. - 1A	1	200R/h (LMRA)	NaI Detector	9/87	7/87	9/87	Ref. GENB-INT-013 (0.7 - 6.7 kg fuel*)
AX115	MUBP Dmain. - 1B	1	200R/h (LMRA)	NaI Detector	10/87	7/87	10/87	Ref. GENB-INT-013 (0.7 kg fuel*)
AX116	MU Tank	1	300mR/h, 300mR/h (LMRA)	NaI or NaCo Detector	8/87	7/87	11/87	Ref. TB-06-00 (Tank: 102 grams fuel*; Relief Valve Pipe: 146 grams fuel*; Tank Discharge Pipe: 6 grams fuel*)
AX117	MUBP Filters (MU-F2ABB and MU-F5ABB)	1	800mR/h	NaCo Detector	9/87		12/87	MU-F2A, 2B probably have fuel MU-F5A, 5B have been changed out and are being rechecked.
AX118	Spent Fuel Coolers	3	5mR/h					
AX119	Spent Fuel Dmain	3	160mR/h					

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**APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)**

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
AX120	Spent Fuel Filters	3	10mR/h					
AX121	Elevator Shaft	3	<0.2 mR/h					
AX122	North Stairwell	3	<2.5 mR/h					
AX123	Access Area	3	<2.5 mR/h					
AX124	Concent. Liquid Waste Pump	2	10mR/h	HpGe Detector or Documentation			...	
AX125	Waste Gas Decay Tank - 1B	3	<2.5 mR/h				...	
AX126	Waste Gas Filter Room	3	<2.5 mR/h				...	
AX127	Waste Gas Decay Tank - 1A	3	<2.5 mR/h					
AX128	Valve and Instrument Room	3	<2.5 mR/h					
AX129	Deborating Demin - 1B	2	<2.5 mR/h	Documentation			...	Unlikely
AX130	Deborating Demin - 1A	2	<2.5 mR/h	Documentation			...	Unlikely
AX131	Misc. Waste Tank (MDL-72)	2	200mR/h (LHRA)	Hal or HpGe Detector	7/87	5/87	6/87	Not surveyed, standpipe drained
AX132	Corridor between Unit 1 and Unit 2	3	<2.5mR/h					
AX133	South Stairwell	3	<2.5mR/h					
AX134	Misc. Waste Tank Pumps	2	20mR/h	HpGe Detector			10/87	Not surveyed
AX135	Radwaste Disposal Control Panel	3	<2.5mR/h					
AX201	North Stairwell	3	<2.5mR/h					

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APPENDIX I - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
AX202	Elevator Shaft	3	0.2mR/h					
AX203	4160V Switchgear - 2-1E	3	<2.5mR/h					
AX204	4160V Switchgear - 2-2E	3	<2.5mR/h					
AX205	RB Purge Air Supply and Hy. Cont. Exh.	3	<2.5mR/h					
AX206	RB Purge Exhaust Unit B	3	26mR/h					
AX207	RB Purge Exhaust Unit A	3	60mR/h					
AX208	Aux. Bldg. Exhaust Unit B	3	5mR/h					
AX209	Aux. Bldg. Exhaust Unit A	3	5mR/h					
AX210	FN Bldg. Exhaust Unit B	3	5mR/h					
AX211	FN Bldg. Exhaust Unit A	3	5mR/h					
AX212	Decay Heat Surge Tank & Substation	3	<2.5mR/h					
AX213	Unit Substations & Access Area	3	<2.5mR/h					
AX214	Decon Facility	3	<2.5mR/h					
AX215	FN Bldg. Supply Unit	3	<2.5mR/h					
AX216	Aux. Bldg. Supply Unit	3	<2.5mR/h					
AX217	Access Area	3	<2.5mR/h					
AX218	Concent. Waste Storage Tank Room	2	20mR/h	HeGe Detector	10/87		11/87	

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APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
AX219	Inst. Racks & Atmosph. Monitor	3	<2.5mR/h					
AX220	Caustic Liquids Mixing Area	3	10mR/h					
AX221	Caustic Liquids Mixing Area Corr.	3	10mR/h					
AX222	South Stairwell	3	<2.5mR/h					
AX221	Air Handling Units General Area	3	<2.5mR/h					
AX301	Elevator Shaft and Elevator Machine Room	3	<0.2mR/h					
AX302	North Stairwell	3	<2.5mR/h					
AX401	Elevator and Stairwell Access	3	<2.5mR/h					
AX401	Roof	3	<2.5mR/h					
AX402	Cooling Water Surge Tanks	3	<2.5mR/h					
AX401	Damper Room	3	<2.5mR/h					
AX501	RB Spray Pump - 1A	3	10mR/h					MDL for DH system piping is 1.6 grams* (TB-86-47)
AX502	RB Spray Pump - 1B	3	200mR/h (LHRA)					
AX503	DH Remov. Cooler and Pump - 1A	3	25mR/h (LHRA)					
AX504	DH Remov. Cooler and Pump - 1B	3	60mR/h (LHRA)					

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APPENDIX 1 - Area Classification and SIM Assessment Schedule (Cont'd)

Area**	Description	SIM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SIM Measurement Date	Remarks
FH001	Makeup Section Valve Room	1	<55mR/h	NaI Detector	3/87	1/87	2/87	Pre-Flush status: Gamma Spectr. measurement-264 grams* in corridor (TB-06-06)
FH002	Access Corridor	3	20mR/h, 150mR/h					
FH007a	Makeup Discharge Valve Room	1	1.5mR/h, 500-(LURA)	NaI or HpGe Detector	4/87	1/87	3/87	Pre-Flush status: Gamma Spectr. measurement-8 grams* (TB-06-07)
FH007b	Makeup Discharge Valve Room	1	5R/h, 700R/h-(LURA)	NaI or HpGe Detector	4/87	1/87	5/87	Pre-Flush status: Gamma Spectr. measurement-30 grams* (TB-06-07)
FH004	Westinghouse Valve Room (mini decay heat)	3	15mR/h					
FH005	Mini-Decay Heat Vault	3	3.2mR/h					
FH006	Decay Heat Service Coolers	3	40mR/h					
FH007	Neutral & Reclaimed Boric Acid	3	100mR/h, 300-					
FH008	Neutralizer Tank Pump Rm.	3	50mR/h					
FH009	Neutralizer Tank Room	3	400mR/h (LURA)					
FH010	Reclaimed Boric Acid Tank	3	2mR/h					
FH011	Reclaimed Boric Acid Pump	3	10mR/h					
FH012	Neutralizer - Tank Fillers	3	800mR/h					
FH013	Oil Drum Storage Area	3	<0.2mR/h					

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APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
FH114	Annulus	2	250mR, 300mR (LMRA)	NaI Detector			1988	Gamma Spectr. measurement scheduled
FH101	NDLP Valve Room	1	400mR (LMRA)	NaI Detector	11/86		4/87	Pre-Flush status: Block orifice removed - 125 grams <sup>m</sup> in orifice; 575 grams <sup>m</sup> in remainder of cubicle (TB-86-21)
FH102	East Corridor	3	20mR					
FH103	Sample Room	3	10mR					
FH104	West Corridor	3	<2.5mR					
FH105	Model Room A	3	10mR					
FH106	Monitor Banks & Sample Sink Area	2	10mR					Unlikely
FH107	Trash Compactor Area	3	<2mR					
FH108	Trash Bag	3	<2.5mR					
FH109	Spent Fuel Pool A	2	<2.5mR					Possible fuel fines from canisters.
FH110	SBS Spent Fuel Pool	2						Possible fuel fines from canisters.
FH111	Fuel Cash Storage	2						Possible fuel fines from canisters.
FH112	Annulus	2	50mR				12/87	Gamma Spectr. scheduled
FH201	East Corridor	3	10mR					
FH202	West Corridor	3	<1.0mR					
FH203	Surge Tank Area	3						
FH204	SPE Area	3	<2.5mR					
FH205	Annulus	3	120mR					

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APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Current Radiation Levels (per hour)	Assessment Method	Surface Decon Date	System Flush Date	SNM Measurement Date	Remarks
FH301	Upper Spent Fuel Pool Area	3						
FH302	SDS Operating Area	2	5mR/h	MoGe Detector or Documentation				
FH101	Upper SPC Area	3	<2.5mR/h					
FH104	Annulus	3	20mR/h					
FH105	Spent Fuel Pool Access	3	<2.5mR/h					

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APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Assessment Method	Estimated Defueling Date	Estimated SNM Measurement Date	Controlling Factors	Remarks
RB01	Leddown Coolers Cubicle	1	NaI Detector	Not to be defueled	Utilize DPA measurements	OQA acceptance of previous DPA measurements	TB-86-26 (<2.3 kg of fuel*)
RB02	Reactor Building Sump	1	Sampling/NaI/HpGe	Not to be defueled	GENB-042	OQA acceptance of previous DPA measurement	GENB-042
RB03	Reactor Coolant Drain Tank Cubicle	1	Video Insp.	Not to be defueled	GENB-042	OQA acceptance of previous DPA measurement	GENB-042 (Tank: <0.1 kg of fuel*)
RB04	Reactor Building Basement (Floor)	1	Sampling/NaI/HpGe	6/87 Desludging	Utilize previous DPA measurement	OQA acceptance of DPA measurement package	TB-86-83, TB-85-88, TB-86-30, TB-86-36 (<3.2 kg of fuel*)
RB05	Under Reactor Vessel	1	TBD	TBD	TBD	Severe access and dose rate problems	TB-86-25
RB06	Leddown Line	2	TBD	TBD	TBD	Severe access and dose rate problems	
RB11	Decay Heat Drop Line	2	Video Insp. & Sampling	TBD	2/87	Requires RCS visibility and defueling window for video inspection	
RB12	Drain Stubs (J-Logs and Steam Generators)	1	Video Insp. & Sampling	Not to be defueled	TBD	Will be performed by extrapolating sample data to drain stub volumes	
RB21	Reactor Coolant Pumps	2	Video Insp. & Sampling	TBD	2/87	Requires RCS visibility and defueling window	
RB22	Horizontal RCS Piping	1	Video Insp. & Sampling	TBD	2/87	Requires RCS visibility and defueling window	

Title  
SNM Accountability Plan

**GE Nuclear**

TH-2 Unit  
Policy/Plan

Revision No.  
0-00

Number  
4000-PLN-4420.02

EI-10

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18/87

Title

SNM Accountability Plan

Revision No.  
0-00

APPENDIX 1 - Area Classification and SNM Assessment Schedule (Cont'd)

Area**	Description	SNM Category	Assessment Method	Estimated Defueling Date	Estimated SNM Measurement Date	Controlling Factors	Remarks
RB23	Reactor Vessel	1	NaI/Video/ Sampling	12/87	6/88	Requires comple- tion of RV defuel- ing and completion of engineering for RV SNM measurement	
RB31	Pressurizer	1	Video Insp.	3/87	3/87		TB-85-89, TB-85-18a, TB-86-02 (<1.2 kg of fuel*)
RB32	Steam Generators		Video/SSTR's/ or Detector				TB-84-85, TB-85-40, TB-86-10, TB-86-23, TB-86-24, TB-86-37, TB-86-38, TB-86-44
	Upper Tube Sheet and Tube Blockages	1		4/87	7/87	Only necessary if SNM is found in lower OTSG regions	(OTSG-A: <43 kg of fuel*; OTSG-B: <74 kg of fuel*)
	OTSG Tube Surfaces	1	Cylindrical Detector	N/A	4/87	Delivery of cylindrical alpha detector	
RB33	Core Flood Tanks - A&B and Drain Lines	2	"A": TBD "B": NaI	TBD	"A": TBD "B": Utilize previous DPA measurement	Possible access and dose rate problems	TB-85-87 ("B" - Core Flood Tank: Drain Line - <120 grams of fuel*; Check Valve - <10 grams of fuel*)
RB34	Incore Guide Tubes	2	NaI Detector				
RB35	Pipeman	1	TBD	TBD	7/87	Dependent upon defueling decisions	TB-84-87
RB36	Reactor Vessel Hood	1	NaI Detector	N/A	8/87		
RB37	Reactor Coolant Hot Legs	1	Documentation (films)	TBD	TBD		
RB38	Pressurizer Surge Line	1	NaI Detector				TB-85-89 (<200 grams of fuel*)
RB39	Pressurizer Spray Line	2	NaI Detector	1/87	TBD	High Area Radiation Dose Rates	
RB40	Fuel Transfer Canal	2					

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GPU Nuclear Corporation  
Post Office Box 480  
Route 441 South  
Middletown, Pennsylvania 17057-0191  
717 944-7621  
TELEX 84-2386  
Writer's Direct Dial Number:

(717) 948-8461

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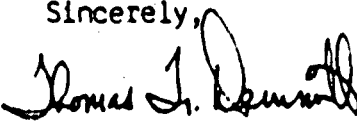
Dr. W. D. Travers -Director  
TMI-2 Cleanup Project Directorate  
US Nuclear Regulatory Commission  
c/o Three Mile Island Nuclear Station  
Middletown, PA 17057

Dear Dr. Travers:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)  
Operating License No. DPR-73  
Docket No. 50-320  
Special Nuclear Material Accountability Plan

At a meeting with the NRC TMICPD on Tuesday, December 23, 1986, GPU Nuclear committed to provide the NRC TMICPD with a docketed copy of the approved GPU Nuclear Special Nuclear Material (SNM) Accountability Plan. Accordingly, attached for your information is a copy of Procedure 4000-PLN-4420.02, Revision 0-00, SNM Accountability Plan," dated April 3, 1987.

Sincerely,

  
F. R. Standerfer  
Director, TMI-2

FRS/RDW/eml

Attachment

A001  
4/11



Buckley, John | Friday, February 10, 2012

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System Properties

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Minor Version:	0
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Size:	1 KB
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THE  
ACCIDENT GENERATED WATER (AGW)  
DISPOSAL  
COMPLETION REPORT

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# AGW COMPLETION REPORT

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## AGW COMPLETION REPORT

### 1.0 Introduction

This document demonstrates that the intent of the TMI-2 program to dispose of Accident Generated Water (AGW) by evaporation has been completed and no continuing unique limitations should be imposed on the processing and disposal of residual water.<sup>1</sup>

### 1.1 Background

The TMI-2 accident resulted in the radioactive contamination of large volumes of water. Direct releases of reactor coolant during the accident filled the Reactor Building basement to a depth of about three and one-half feet. Following the accident, water was added to this inventory by primary coolant leakage and inleakage of river water through the Reactor Building air coolers. A Programmatic Environmental Impact Statement (PEIS), completed in March, 1981, (Reference 1) stated that a decision on the ultimate disposal of the AGW could be deferred until after the water had been processed. Consequently, the NRC issued a Policy Statement on April 27, 1981 (Reference 2), which included a requirement that any future proposal for disposition of processed AGW shall be referred to the Commission and the Commission would reserve, unto itself, the right of approval. This resulted in an amendment to the TMI-2 plant Technical Specifications, which prohibited discharge of AGW without prior NRC approval.

In accordance with the TMI-2 Technical Specification, AGW is defined as:

- "(a) Water that existed in the TMI-2 Auxiliary, Fuel Handling and Containment buildings, including the primary system, as of October 16, 1979, with the exception of water which as a result of decontamination operations becomes commingled with non-accident-generated water such that the commingled water has a tritium content of 0.025  $\mu\text{Ci/ml}$  or less before processing;

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<sup>1</sup>Although AGW evaporation is not scheduled for completion until the 3rd quarter of 1993, this document is written from the perspective that evaporation is complete, i.e., the tanks and system piping are drained to the levels detailed in this report. GPU Nuclear will report to the NRC by letter when the evaporation of AGW has been completed.

- (b) Water that has a total activity of greater than one  $\mu\text{Ci/ml}$  prior to processing except where such water is originally non-accident water and becomes contaminated by use in cleanup;
- (c) Water that contains greater than 0.025  $\mu\text{Ci/ml}$  of tritium before processing."

EPICOR II began processing of AGW from the Auxiliary Building in October, 1979. Processing of AGW from the Reactor Building using the Submerged Demineralizer System (SDS) was initiated in mid-1981. Since 1979, the total inventory of AGW increased to approximately 2.3 million gallons due to continued additions as a result of defueling and decontamination activities, condensation from the Reactor Building air coolers, rain and ground water leakage and leakage from systems containing demineralized water.<sup>2</sup>

In 1987, the NRC completed Supplement No. 2 to the PEIS which addressed the disposal of AGW (Reference 3). In reviewing GPU Nuclear's proposal to dispose of AGW by forced evaporation to the atmosphere (References 4 and 5), the NRC evaluated nine alternatives including long-term and short-term discharge to the Susquehanna River.

The NRC concluded that no alternative was clearly preferable to GPUN's proposal for evaporation. While the quantitative estimates of potential impacts were found to vary for some of the alternatives, the differences were not judged to be sufficiently large to allow for either identification of a clearly preferable alternative or rejection of any of the 9 options evaluated. Following the completion of a contested hearing before the Atomic Safety Licensing Board, the NRC approved the GPU Nuclear plan to evaporate AGW (Reference 6) as an acceptable disposal plan. The TMI-2 Recovery Technical Specifications were revised to remove the prohibition on disposal of AGW and to allow disposal of AGW in accordance with NRC approved procedures.

Prior to the initiation of AGW disposal operations, most of the AGW had been processed to very low levels of radionuclide contamination; this AGW is commonly referred to as processed water. Processed water was recycled for use in cleanup activities and was subsequently reprocessed.

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<sup>2</sup>e.g., Nuclear Service Closed Cooling Water, Demineralized Water

In January, 1991 GPU Nuclear began disposal of AGW via the Processed Water Disposal System (PWDS). AGW disposal was completed during 1993. The PWDS disposed of an estimated 99% of the initial pre-processing volume of 2.3 million gallons. The residual volume is estimated to be approximately 18,500 gallons; < 1% of the pre-disposal volume.

The GPU Nuclear method for disposal of AGW, as approved by the NRC, utilized a two-cycle evaporator/vaporizer system to process the water through a closed cycle evaporator, reheat the purified distillate, and discharge the vapor to the atmosphere. This process removed essentially all of the soluble material and particulate contamination (i.e., >99.9%) which was concentrated in the evaporator bottoms, collected and further concentrated to a dry solid that was shipped for disposal by burial at a commercial low level radioactive waste facility. The remaining radioactivity, including the tritium, was released as vapor. The effluent vapor discharge was monitored. Water that required additional processing to reduce its radionuclide concentrations prior to disposal was processed by ion exchange, filtration, or distillation. This pre-processing was accomplished using the existing EPICOR II System or by operating the evaporator in a closed cycle mode (i.e., no vapor release to the atmosphere), or both.

Selection of evaporation as the GPU Nuclear preferred method of AGW disposal was based in large measure on the public perception that AGW posed a unique hazard to public health and safety because it was related to the 1979 accident at TMI-2. The technical merits of the various disposal options, including discharge to the Susquehanna River and evaporation, and their potential environmental impacts were judged to be comparable and were not at issue in the selection process.

Having collected and disposed of the AGW by the evaporative process to the extent reasonably achievable consistent with ALARA, it is now proposed that the program to dispose of AGW by evaporation be considered complete and that the residual quantities of water remaining at TMI-2 be subject to no unique processing or disposal restrictions. GPU Nuclear believes that the residual TMI-2 water should be disposed in accordance with the same limits and conditions imposed on all other TMI waste water, i.e., discharged to the Susquehanna River in accordance with existing license condition and liquid discharge requirements.

## 1.2 Report Organization

Section 2 of this report discusses expected volumes, locations and isotopic content of the residual water. This section also addresses the extremely small risk of residual water discharge to the Susquehanna River.

Section 3 summarizes the evaporation process and the effort to remove and dispose of as much AGW as reasonably achievable from within TMI-2.

Section 4 summarizes the planned disposition of the residual water at TMI-2 including dilution and discharge of some volumes and long-term, natural, in-situ evaporation for other volumes.

Section 5 presents the conclusions reached by GPU Nuclear regarding the ultimate termination of the AGW evaporation process and acceptability of removing any special restrictions on the processing and disposal of the residual water.



## 2.0 Residual Water Description

When AGW processing is completed, a very small amount of contaminated water will remain in TMI-2 systems, piping and building sumps. Essentially all of this water is, by definition<sup>1</sup>, not AGW. Therefore, for the purposes of this report, water remaining at TMI-2 after the completion of AGW processing will be referred to as residual water for which AGW disposal requirements are not applicable. The residual water is predominantly contained in the bottoms of tanks and sumps, and in piping from which additional water cannot be removed practically.

### 2.1 Location, Volumes and Content

Table 1 compares the locations and quantities of the residual water<sup>4</sup> in tanks to the initial volumes of AGW reported in January, 1986. In some locations, the final volumes are conservative estimates because of the physical configuration of the tank or level instrument inaccuracy at the near-empty levels.

Table 2 provides the results of the system-by-system draindown to remove the maximum amount of AGW from system piping. The total estimate of residual water in the system piping is < 6550 gallons, isolated in a number of discrete locations and very small volumes.

The residual water volumes reported could not be drained because of physical or mechanical impediments, such as elevation, physical location or component failure. ALARA considerations also prevailed<sup>5</sup>.

Calculations were based on known physical parameters including tank/vessel size, component capacities, water levels and pipe dimensions.

Estimated residual water volumes are based on a summation of calculated volumes in non-accessible components/piping that could not be verified as completely drained (e.g., small puddles of water remaining in horizontal piping, traps, instrumentation lines and pump casings). It has been confirmed that most

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<sup>1</sup>Analysis of residual water samples from the locations where the majority of the residual water is located, i.e., has shown that over 85% of the water in these locations does not meet the definition of AGW (Table 3).

<sup>4</sup>The residual water volumes listed in Tables 1 and Table 2 are conservative estimates based on the known physical dimensions of the component (tank or piping) or structure (sump) and, where possible, definitive level indication.

<sup>5</sup>In some systems (e.g. Pressurizer surge line drain), low-point drain valves were inoperable and the dose required to repair the valve and drain the residual water was not justifiable from an ALARA perspective.

systems are not completely drained. In some cases the difference in total calculated water volume versus the volume actually drained/collected was also used as an aid in determining estimated potential residual water remaining.

## 2.2 System Draining

Due to the variety of systems present in the plant, individual drain procedures were prepared for each system or, in some cases, part of the system to be drained. In order to ensure retrieval and processing of the maximum available quantity of AGW, existing TMI-2 plant systems containing AGW were drained to the extent reasonably achievable by gravity draining, air blowdown and/or by using existing or temporary pumps. Gravity draining was accomplished by isolating systems with closed valves, opening system high point vents and then draining the system via low drain points. Some piping was flushed with demineralized water to maximize AGW removal.

AGW removed from system piping and tanks in the Reactor Building (RB) was transferred to the Reactor Coolant Bleed Tanks and then sent to EPICOR II for processing prior to evaporation. AGW in the RB basement was removed using a submersible pump and processed through the Submerged Demineralizer System (SDS) prior to transfer to EPICOR II for processing. When the level of water in the RB basement became too low for collection by the submersible pump, AGW was drained from the RB sump to the Auxiliary Building where it was routed to EPICOR II via the Miscellaneous Waste Holdup Tank (MWHT).

AGW was removed from system piping and tanks in the Auxiliary Building by pumping directly to EPICOR II or draining the piping and tanks to the Auxiliary Building sump. Auxiliary Building sump water was transferred to EPICOR II via the MWHT.

AGW was removed from system piping and tanks in the Fuel Handling Building by either draining the piping to the Reactor Coolant Bleed Tanks in the Auxiliary Building or direct transfer to EPICOR II or the Auxiliary Building sump.

The final estimate of the total quantity of residual water that will remain after the completion of AGW processing is no more than 18,500 gallons. This volume is < 1% of the AGW inventory that existed in December 1990, i.e., prior to PWDS operation.

Table 3 lists the activity concentration in  $\mu\text{Ci/ml}$  of the most significant radioactive isotopes in the MWHT, the Auxiliary Building Sump, the "A" Concentrated Drain Tank, and the RB Sump. This table indicates only a few locations of high activity water remain for volumes of greater than several hundred gallons.

### 2.3 Hazard Analysis

The potential hazards of the storage and future processing of residual water at TMI-2 were reviewed. This review determined that the spill of 19,000 gallons of processed water during a transfer of the water from the Chemical Cleaning Building (EPICOR II) to a release pathway for ultimate discharge to the Susquehanna River is considered to be a maximum plausible accident involving the inadvertent release of residual water after the completion of AGW evaporation. For the purposes of this analysis, the spill is assumed to occur during the transfer of the water from the Chemical Cleaning Building to a discharge pathway, after the residual water has been processed. The residual water is assumed to be spilled onto the ground surface external to site buildings. The radionuclide concentrations in the spilled water are assumed to be equivalent to that of base case water (reference 4). A volume of 19,000 gallons was chosen because that is the capacity of the TMI-2 MWHT, the tank from which supply batches of residual water will be transferred to the Chemical Cleaning Building for processing after the completion of AGW evaporation. A portion of the residual water evaporates and delivers dose to the maximally exposed individual (MEI) via the acute inhalation pathway. The remainder of the water is absorbed into the ground and travels via the ground to the river.

The dose to the MEI from the postulated spill of 19,000 gallons of processed water is bounded by previous analysis performed by the NRC of the spill of 600,000 gallons of processed water (reference 4). The NRC concluded that a spill of 600,000 gallons (i.e., the volume of a Processed Water Storage Tank) of processed AGW was bounded by the instantaneous release of the entire 2.3 million gallon volume of AGW to the Susquehanna River. NRC analysis of the dose resulting from the release of the entire 2.3 million gallons of AGW to the Susquehanna resulted in a bone dose of 3 mrem and a whole body dose of 0.4 mrem to the MEI. Thus, it can be concluded that the spill of 19,000 gallons of processed residual water will result in a dose to the MEI that is a small fraction of the 10 CFR 50 Appendix I guidelines (no more than 10 mrem to any organ and no more than 3 mrem whole body).

A spill of processed water was chosen for this hazard analysis instead of a spill of unprocessed RB sump water because there is no credible spill path for unprocessed water to be released. RB sump water is the most radioactive residual water that will remain after the completion of AGW evaporation. During the

remainder of Mode 3 and PDMS it is credible that residual water in the RB sump may accumulate to a level that requires processing and disposal. RB Sump water will be drained to the Building Spray sump and then pumped to the MWHT via the Auxiliary Building sump. The MWHT will be the holding tank for residual water awaiting processing.

An uncontrolled release of unprocessed RB Sump water is not a credible event. Any spillage during the transfer of RB Sump water from the Reactor Building to the MWHT will flow into either the Building Spray sump or the Auxiliary Building sump. Spillage that occurs during the transfer of the RB Sump water between the MWHT and the external wall of the Auxiliary Building will flow into the Auxiliary Building sump via the floor drain system. Spillage from the piping between the Auxiliary Building and the Chemical Cleaning Building has been determined by the NRC to be an incredible event (References 11 and 12) because the supply pipe from the MWHT to the Chemical Cleaning Building is enclosed by a 4" guard pipe which is embedded in concrete. A spill of unprocessed RB Sump water from the Chemical Cleaning Building to the external ground is not a credible event because the Chemical Cleaning Building was constructed as a concrete "bathtub" capable of retaining the entire contents of the 2 storage tanks (approximate total volume of 200,000 gallons) located within the building. Therefore, the only credible release of residual water as a result of processing and handling could occur when processed water is transferred from the Chemical Cleaning Building to a release pathway.

### 3.0 Evaporation Program

An estimated 2.3 million gallons of processed TMI-2 AGW accumulated by the end of the TMI-2 Clean Up Program in 1990 (Reference 6). The AGW required disposition in accordance with TMI-2 Recovery Technical Specification 3.9.13. Prior NRC approval of the disposal of this water by evaporation was required and received (Reference 7). As stated in section 1 of this report, AGW disposal via the Processed Water Disposal System (PWDS) was initiated in January, 1991.

#### 3.1 Process Summary

The PWDS disposed of the AGW via a two-stage evaporation process. The PWDS consists of: (1) a vapor recompression distillation unit (main evaporator) that distilled the processed water in a closed cycle and collected the purified distillate for subsequent release by vaporization; (2) an auxiliary evaporator that further concentrated the bottoms from the main evaporator; (3) a flash vaporizer unit that heated and vaporized the purified distillate from the main evaporator and released the vapor to the atmosphere in a controlled and monitored manner; (4) a waste dryer that further evaporated water from the concentrated waste and produced a dry solid; and (5) a packaging system that prepared the dry solid waste in containers acceptable for shipment and burial in a commercial low level radioactive waste disposal site.

The influent quality was controlled to ensure effluent limits were achieved. The purified distillate released to the environment via the vaporizer contained a level of radioactive contaminants which did not exceed 1/1000 of the concentration of dissolved radioactive contaminants in "Base Case" water (see Reference 6). The level of contaminants released in the vapor also was maintained sufficiently low to ensure minimal environmental impact.

At least 99.9 percent of the dissolved radioactive contaminants contained in the base case evaporator influent were collected as dry solid waste. This waste was packaged onsite and transported for burial in a commercial radioactive waste disposal facility. The waste form was suitable for transportation and burial in accordance with the federal Department of Transportation and NRC regulations. GPUN chose to process the waste to a form that met the transportation requirements for Low Specific Activity (LSA) radioactive material. In addition, it conformed to the burial requirements for Class A waste. In general, LSA and Class A waste forms constitute the lowest level of radioactive waste material which originates from commercial nuclear power plants and is regulated for purposes of transportation and disposal.

At the outset, the PWDS disposed of water stored in various tanks in the plant. Most of the water disposed by the PWDS was processed through the Submerged Demineralizer System (SDS) and/or EPICOR II prior to PWDS processing. Some of the 2.3 million gallon inventory (e.g., the AGW in the Reactor Coolant System) required additional preprocessing before being disposed by the evaporator. This water was processed by the PWDS in the closed cycle decoupled mode<sup>6</sup> prior to being processed and disposed by the PWDS in a coupled mode. In all cases, the PWDS was operated in a manner such that the PEIS projections (Reference 4) of environmental impact were not exceeded.

### 3.2 Water Processed

The evaporation process was used to process and dispose of >99% of the 2.3 million gallons of AGW. The residual water, estimated volume of 18,500 gallons (see Tables 1 and 2), is distributed in numerous locations with no single location containing as much as 6,000 gallons.

### 3.3 Residual Water

While all reasonable efforts have been expended to collect and dispose of AGW, some residual water remains in tanks and piping volumes (Tables 1 and 2). In specific cases, additional effort was undertaken to open systems by removing components or by wet vacuuming where practical and consistent with sound ALARA practice. However, in many cases it was not considered ALARA to attempt to enter high radiation areas to gain access to additional small quantities in tanks and pipes, especially in the Reactor Building. For these reasons, residual water remains as described in Section 2.0.

The relatively small quantities of residual water which remain in numerous locations throughout the TMI-2 facility do not pose any threat to health and safety. This residual water is well contained and cannot effect the long-term safety and integrity of the facility. In fact, for the largest volumes of residual water, over 85% is not categorized as AGW, as defined in Section 1.17 of the Appendix A Technical Specifications. Because of the radiation fields and lack of accessibility at the residual water locations, additional radiation exposures to workers cannot be justified for the negligible benefit that might result from removal and disposition of this water. In accordance with the fundamental principles of good radiation control practices, occupational exposures to radiation should be kept as low as reasonably achievable (ALARA). It is prudent to forego any activity which does not provide a benefit commensurate with occupational

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<sup>6</sup>When the PWDS was operated in the closed cycle decoupled mode, the main evaporator distilled the influent liquid which was then condensed and pumped to plant storage tanks. The PWDS did not process AGW through the vaporizer for disposal when operated in the decoupled mode.

exposures required to complete that activity. GPU Nuclear is convinced the point has been reached such that further water removal and disposal as AGW is an activity that clearly cannot be justified because of the occupational exposures required for negligible benefit.

Additional small volumes not drained include pipe runs with numerous instrument taps; seismic pipe (i.e., welded); pipe sections in high dose areas where access is inconsistent with sound ALARA practice; pipe sections in high contamination areas; and systems designed not to be drained.

Much of the AGW removed during final system draining was routed for processing through the Auxiliary Building sump to the MWHT. Following the completion of all AGW removal, both the MWHT and the Auxiliary Building<sup>7</sup> sump underwent a final fill, recirculation and flush evolution to maximize the removal of residual radioactivity. The residual water in each of these areas no longer meets the criteria for AGW after the fill, recirculation and flush cycles.

After the TMI-2 accident, approximately 640,000 gallons of water flooded the RB basement (Reference 10). This water was pumped to the SDS Tank Farm in 50,000 gallon batches using a sump pump. The Tank Farm water was processed in the Fuel Handling Building by the SDS.

After the initial AGW volume was removed to the extent achievable from the RB basement, the basement was partially refilled with processed water to reduce the dose rates on the upper elevations of the RB. The RB basement water volume was lowered when the RCS was depressurized and transitioned from the pressure control mode to the level control mode in preparation for defueling. Because of safety considerations,<sup>8</sup> the RB basement water volume was maintained below 70,000 gallons throughout the remainder of the cleanup program. Excess water was removed and processed by the SDS. The RB basement was drained during

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<sup>7</sup>The water level in the Auxiliary Building sump is maintained above the floor drain discharge piping penetrations to limit the recontamination of AFHB cubicles via the floor drain system.

<sup>8</sup>After the March, 1979 accident, the TMI-2 reactor coolant was maintained with a relatively high soluble concentration of 3000-5000 ppm Boron. This ensured subcriticality of the core was maintained for even the most reactive core debris geometry. In order to be certain that, in the event an unisolable leak occurred in the Reactor Coolant System, the RB basement water could be safely recirculated through the Reactor Vessel, a limit of approximately 70,000 gallons was placed on the RB basement water volume. Assuming there was no dissolved boron in the RB basement water, an unisolable RCS leak would flow into the RB basement, mix with the basement water and be available for reflood of the Reactor Vessel. Because the Borated Water Storage Tank and the RCS together contained approximately 460,000 gallons of water with over 4950 ppm boron, the mixing of 70,000 gallons of unborated water would result in a RCS boron concentration of over 4000 ppm, significantly above the 3500 ppm post-accident RCS boron concentration.

the mid-1980's using a submersible pump that was placed in the incore pipe chase<sup>9</sup>. RB basement water was diluted during the summer months by the addition of over 10,000 gallons a month of non-AGW water via condensation<sup>10</sup>. Finally in 1992, the RB basement sump was subjected to an additional draindown via the 18" Decay Heat RB sump suction lines that connect to the RB Building Spray suction line<sup>11</sup>. An estimated residual volume of <1,500 gallons remained in the RB sump after this draindown. The radioactivity content<sup>12</sup> of the remaining water is listed in Table 3.

The Reactor Building sump contains the largest volume of residual water with a significant amount of radioactivity (Table 3). The RB sump is constructed such that the sump is divided into two chambers separated by a weir. The RB floor drains feed into one of the chambers which, when full, overflows into the other chamber which can be level-monitored, sampled, and drained via the Decay Heat sump suction line to the RB Building Spray recirculation suction line.

The RB sump residual water is the remnant water that remains on the non-drained side of the weir in the sump after a series of decontamination and basement water processing evolutions that removed essentially all of the AGW. The residual water currently in the RB sump has a tritium concentration of less than 0.025  $\mu\text{Ci/ml}$  and a total activity of approximately 4  $\mu\text{Ci/ml}$ . Essentially all of the water currently in the RB sump was originally non-accident water that became contaminated as a result of cleanup operations. A comparison of the reported concentrations of H-3 (tritium), Cs-137 and Sr-90 in RB basement AGW immediately after the March, 1979 accident (Reference 9), to the current concentration of these isotopes is listed below.

---

<sup>9</sup>This draindown left very little water on the RB basement floor. A rough estimate of the volume of water remaining in the RB basement after this draindown including the RB sump was less than 10,000 gallons.

<sup>10</sup>An air conditioning system called the RB Air Chiller System was installed at TMI prior to the initiation of defueling. The RB Air Chiller System was "piggy-backed" onto the RB Normal Air Cooler System. These combined systems were capable of maintaining the RB air temperature at an ambient temperature of 65 degrees F or less. The operation of the RB Air Chiller System resulted in a significant increase of water flowing to the RB basement because the cooler air temperatures caused the moisture in the air to condense and flow into the RB basement and RB sump. TMI-2 Liquid Radwaste Management reports (References 9 and 10) indicate that a monthly addition of 10,000 gallons was routine during chiller operations. Assuming that the chillers were operated for effectively 4 months a year from 1985 through 1991, operation of the chillers added as much as 240,000 gallons of non-AGW to the RB basement.

<sup>11</sup>The RB sump pumps (WDL P-2A and P-2B) are in an undetermined condition. They have not been refurbished since the March, 1979 accident. The RB sump was drained using the Decay Heat sump suction line which connects to the RB Spray System recirculation suction line.

<sup>12</sup>The wet side of the RB sump cannot be directly sampled. The reported concentrations of radioactivity are based upon samples of water drained from the dry side of the RB sump.



Isotope	August, 1979	September, 1992
H-3	1.03 $\mu\text{Ci/ml}$	0.018 $\mu\text{Ci/ml}$
Cs-137	176.3 $\mu\text{Ci/ml}$	3.2 $\mu\text{Ci/ml}$
Sr-90	2.81 $\mu\text{Ci/ml}$	0.45 $\mu\text{Ci/ml}$

The continual addition of non-AGW to the RB basement sump, via condensation and draindown, diluted the AGW present in the sump to the extent that the tritium concentration is less than 0.025  $\mu\text{Ci/ml}$ .

#### 4.0 Disposition of Residual Water

All reasonable and practical efforts have been completed consistent with sound ALARA practices to collect and process the AGW. The residual water represents < 1% of the original volume and is distributed in small volumes throughout the plant. No further efforts will be directed to remove and evaporate this small amount of residual water. The evaporator has been retired. The ultimate disposition of the residual water is anticipated as follows:

#### 4.1 Periodic Dilution and Discharge

Some residual water volumes may be diluted during PDMS due to building atmospheric condensation, rain/groundwater leakage and limited decontamination maintenance efforts. Liquid radwaste management systems are being maintained operable to deal with this water buildup and include both the Rad Waste Disposal Miscellaneous Liquid System and the Sump Pump Discharge and Drainage System. Portions of these systems are being maintained operable to prevent localized flooding as well as to provide proper disposal of liquid effluents.

As part of the Waste Disposal Liquid (WDL) System, portions of the Miscellaneous Liquid System remain operational. The operational status of the WDL provides assurance that significant quantities of liquid wastes will not accumulate in an uncontrolled manner in the Auxiliary Building and Containment. The WDL System achieves its objective by meeting the following criteria:

- a. Existing sumps in the Auxiliary Building and Containment will be monitored and pumped, as required.
- b. Tie-ins to the EPICOR II or other appropriate processing system will be maintained so that accumulated liquids can be processed, as necessary.
- c. Liquid storage capabilities will be maintained for accumulation of leakage and residual water until sufficient quantities are available for batch processing.
- d. If required during PDMS, the operable portions of TMI-2 WDL System can receive liquids from the AFHB, assorted equipment in these buildings, and from the RB sump. This system has the capability to retain waste liquids to allow for radioactive decay, sampling, filtration or transfer for processing and/or disposal.

Because a majority of plant systems have been deactivated, drained and placed in a layup condition, there are a limited number of activities that can generate liquid waste. Liquid waste in the remaining operable systems and accumulated leakage will be adequately handled by periodic batch processing using the operational portions of the WDL System through EPICOR II or an equivalent system; discharge will be via approved pathways in accordance with existing liquid release limits. This ensures minimum exposure to plant personnel and minimizes releases to the environment in accordance with 10 CFR 20 and 10 CFR 50 Appendix I.

Water entering the active sumps from floor drains in some areas of the plant is generally not contaminated. However, these sumps, within the Turbine Building, Control Building Area, Control and Service Buildings and Tendon Access Gallery, are equipped with recirculation and sample lines to allow sampling for radioactivity.

Monitoring of the levels in the various sumps by remote means and/or visual inspections ensures that accumulated leakage is transferred for processing and disposal in a timely manner before sumps overflow potentially contaminated water onto building basement floors. Sampling quantifies radioactive content and ensures proper waste stream processing. Therefore, the various building sump sampling and discharge capabilities ensure liquid waste streams generated during PDMS are appropriately transferred for ultimate processing and disposal.

Maintaining the various building sumps operational assures that water buildup does not cause adverse localized flooding. These sumps will contain water that either meets or exceeds release criteria. Radioactive water that exceeds release criteria will be routed for processing, then re-sampled and analyzed. Processed water that meets release criteria will be discharged via approved pathways. Water that meets release criteria and does not require processing will be routed to the IWTS and released in accordance with 10 CFR 20 and NPDES regulations via approved pathways.

#### 4.2 In-Situ Evaporation

The majority of the remaining locations containing the residual water described in Section 2.1 will not be accessed during PDMS; some of that water will evaporate by natural processes. The off-site dose consequences are considered inconsequential because the residual water is less than 1% of the original volume which was forcibly evaporated. In addition, the majority of the radioactivity in the residual water will be left behind after natural evaporation occurs, much like salt is left when seawater evaporates.

## 5.0 Conclusions

The vast majority of AGW has been disposed by evaporation in accordance with the TMI-2 Recovery Technical Specifications. The evaporation process was selected by GPU Nuclear from among several environmentally safe options, including discharge to the Susquehanna River, in order to minimize the public reaction based on a perception of the existence of a unique hazard associated with AGW. Over 99% of the AGW has been disposed of; the intent of the GPU Nuclear proposal to evaporate the AGW has been met within the limits of reasonableness, practicality and ALARA. However, GPU Nuclear is faced with management of a small volume of residual water that cannot be recovered and evaporated practically without unwarranted occupational dose and unreasonable effort.

The small amount of residual water remaining at TMI-2, less than 1% of the original volume, poses no significant impact in terms of offsite radiological exposure, i.e., the worst case offsite release is estimated at less than the annual 10 CFR 50 Appendix I limits for radiation exposure to any organ or to the whole body. GPU Nuclear has pursued every reasonable course in disposing of the bulk of the AGW, reducing residual water volumes to <6,000 gallons in any single location. It is, therefore, concluded that no further effort to collect and evaporate or segregate the residual water is warranted. The remaining locations and volumes of residual water can be managed in accordance with the normal provisions for liquid waste discharge at TMI without preserving any unique requirements for disposal.

AGW disposal is complete. The information presented in this report, provides the basis for concluding that the purpose of the AGW disposal program (i.e., collection and disposal to the extent reasonably achievable consistent with sound ALARA practices) has been met. Further, it is concluded that deletion of Recovery Technical Specification 3.9.13, which establishes the unique effluent limits for AGW, is appropriate. In summary, GPU Nuclear has concluded that it is not appropriate to continue the special processing and disposal of residual TMI-2 waste water or to require unique limitations and conditions for its discharge. Residual water at TMI-2 should be processed and discharged in a manner consistent with existing liquid discharge limits and regulations for TMI. Thus, Sections 1.17, 3.9.13 and 3/4.9.13 should be deleted from the TMI-2 Recovery Technical Specifications.

**TABLE 1**  
**TANK DESCRIPTION, INITIAL AND FINAL WATER VOLUME**

STORAGE LOCATION	DESCRIPTION	01/01/86 VOLUME (GALLONS)	FINAL VOLUME (GALLONS) <sup>1)</sup>
PWST-1	Processed Water Storage Tank No. 1	109,000	0
PWST-2	Processed Water Storage Tank No. 2	480,000	0
BWST	Borated Water Storage Tank	459,000	150
SEP-A	"A" Spent Fuel Pool	205,000	0
SEP-B	"B" Spent Fuel Pool	242,000	0
COT-1A	"A" Condensate Storage Tank	102,000	0
FTC	Fuel Transfer Canal	59,000	0
RC'S	Reactor Coolant System (RV, A&B Steam Generators, Preheaters)	67,000	200
RB Sump	Reactor Bldg. (Containment) Sump/Overflow	43,000	1,400
RCBT-A	"A" Reactor Coolant Bleed Holdup Tank	3,800	0
RCBT-B	"B" Reactor Coolant Bleed Holdup Tank	4,400	0
RCBT-C	"C" Reactor Coolant Bleed Holdup Tank	57,000	0
CC-T-1	PWDS Inf./Eff. (EPICOR off-spec) Tank	21,000	0
CC-T-2	EPICOR Receiving Tank	17,000	0
MWHT	Miscellaneous Waste Holdup Tank	3,700	3,000
WDL-T-9A	"A" Evaporator Condensate Test Tank	5,600	0
WDL-T-9B	"B" Evaporator Condensate Test Tank	2,200	0
WDL-T-8A	"A" Neutralizer Tank	8,700	0
WDL-T-8B	"B" Neutralizer Tank	8,600	0
WDL-T-11A	"A" Contaminated Drain Tank	1,900	1,000
WDL-T-11B	"B" Contaminated Drain Tank	800	0
Aux. Sump	Auxiliary/Fuel Handling Building Sump	5,900	6,000
CCB Sump	Chemical Cleaning Bldg. (EPICOR II) Sump	1,000	200
SDS-T-1A	"A" SDS Monitor Tank	400	0
SDS-T-1B	"B" SDS Monitor Tank	500	0
RB Misc.	Miscellaneous Reactor Building Storage	16,000	0
CWST	Concentrated Waste Storage Tank	6,500	0
SRST-A	"A" Spent Resin Storage Tank	900	0
SRST-B	"B" Spent Resin Storage Tank	300	0
1986 AGW VOLUME		1,932,900 <sup>14</sup>	11,950

<sup>1)</sup>Volumes listed are estimates and are representative of the tanks/vessels only. Residual water volumes in associated system piping, instrument lines, etc. are provided in Table 2. Tanks listed as having "0" gallons of residual water are either empty or have only a very small amount of water.

<sup>14)</sup>The volume of AGW increased from the 1986 volume of 1,932,900 gallons to 2,300,000 gallons by January, 1991 (Reference 6) when the PWDS began operation. The additional water was generated as a result of defueling and decontamination activities and condensation from the Reactor Building air coolers.

TABLE 2

ESTIMATED RESIDUAL WATER INVENTORY  
IN SYSTEM PIPING FOLLOWING DRAINING

SYSTEM	RESIDUAL WATER <sup>11</sup> (GALLONS)
Reactor Coolant System (RCS)	< 800
Decay Heat Removal System (DH)	< 1200
Core Flood System (CF)	< 480
Make-Up Reactor Coolant & Purification System (MU)	< 300
Spent Fuel Cooling System (SF)	< 320
Reactor Building Spray System (BS)	< 70
Radwaste Disposal Reactor Coolant Liquid System (WDL)	< 380
Radwaste Disposal Miscellaneous Liquid System (WDL)	< 1150
Radwaste Disposal Reactor Coolant Leakage Recovery System (WDL)	< 370
Chemical Addition System (CA)	< 30
Sampling Nuclear System (SN)	< 15
Defueling Water Cleanup Reactor Vessel Cleanup System (DWC/RV)	< 200
Defueling Water Cleanup Fuel Transfer Canal/Spent Fuel Pool Cleanup (DWC-FTC/SFP)	< 240
Processed Water Storage and Recycle System (PW)	< 200
Sludge Transfer System (STS)	< 10
Radwaste Disposal Solid System (WDS)	< 60
Auxiliary Building Emergency Liquid Cleanup System (ALC)	< 125
Temporary Nuclear Sampling System (SNS)	< 15
Submerged Demineralizer System (SDS)	< 80
Radwaste Disposal Gas (WDG)	< 5
Reactor Coolant Pump Oil Shield Drain Tanks	< 500
TOTAL FOR SYSTEM PIPING	< 6,550

<sup>11</sup>Residual water estimates were conservatively determined using known physical parameters including water levels, piping isometries, known capacities and measured volumes drained from systems. When the quantity of water in piping could not be determined prior to draining, the piping was assumed to be full. The volume of water reported is the difference between the estimated volume prior to draining and the amount drained.

TABLE 3

## Activity Concentration Summary for Residual Water Storage by Location

Location	H <sup>1</sup>	Ce <sup>140</sup>	Sr <sup>90</sup>	Sb <sup>125</sup>	Cs <sup>134</sup>	Cs <sup>137</sup>	Eu <sup>154</sup>	Gross Alpha	Residual Water Volume
RB Sump <sup>16</sup>	1.8E-2	3.7E-4	4.5E-1	<6.2E-3	1.2E-2	3.2E0	<7.93E-3	<5.0E-5	1400
MWHT <sup>17</sup>	1.62E-2	9.13E-5	6.13E-2	<8.68E-4	6.21E-4	1.91E-1	<1.93E-5	<9.57E-6	3000
Aux. Bldg. Sump <sup>18</sup>	1.62E-2	9.13E-6	6.13E-2	<8.68E-4	6.21E-4	1.91E-1	<1.93E-5	<9.57E-6	6000
Contaminated Drain Tank "A" <sup>19</sup>	9.85E-3	2.95E-6	N/A	<5.94E-6	1.55E-6	5.29E-4	<9.09E-7	N/A	1000

Note: All units for isotopic activity, including gross alpha, are in  $\mu\text{Ci/ml}$ . Water volume is in gallons.

<sup>16</sup>RB sump water radioactivity is inferred from a sample taken June 25, 1992 from the RB sump water drained to the "C" Reactor Coolant Bleed Holdup Tank via the MWHT.

<sup>17</sup>TMI-2 Sample Analysis Summary Sheet dated April 11, 1993 from sample taken on April 9, 1993 from MWHT.

<sup>18</sup>Based on MWHT sample taken on April 9, 1993 after MWHT received transfer from Auxiliary Building sump.

<sup>19</sup>Based on WDL-T11A sample taken on June 25, 1993.

## 6.0 References

1. USNRC, NUREG-0683, "Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting From March 28, 1979 Accident," Three Mile Island Nuclear Station, Unit 2, March, 1981.
2. USNRC, "Statement of Policy Relative to the NRC Programmatic Environmental Impact Statement On the Cleanup of Three Mile Island Unit 2," April 27, 1981.
3. USNRC, NUREG-0683 Supplement No. 2, Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting From March 28, 1979 Accident Three Mile Island Nuclear Station, Unit 2 Final Supplement Dealing with Disposal of Accident Generated Water," June, 1987.
4. Letters, GPUNC to NRC, "Disposal of Processed Water," July 31, 1986 and October 21, 1986.
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6. USNRC to GPUNC, "Issuance of Amendment and Approval of the TER on Processed Water Disposal System," September 11, 1989.
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9. GPUNC TPO/TMI-27, "Reactor Building Basement - History and Present Conditions," November, 1982, Getachuw Worku.
10. USNRC, NUREG-0683, "Draft Programmatic Environmental Impact Statement related to the decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station, Unit 2," July, 1980.
11. USNRC, NUREG-0591, "Environmental Assessment Use of EPICOR-II At Three Mile Island, Unit 2," August 14, 1979.



Buckley, John | Friday, February 10, 2012

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**GPU Nuclear Corporation**  
Post Office Box 480  
Route 441 South  
Middletown, Pennsylvania 17057-0191  
717 944-7621  
TELEX 84-2386  
Writer's Direct Dial Number:

(717) 948-8400

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US Nuclear Regulatory Commission  
Attn: Document Control Desk  
Washington, DC 20555

Three Mile Island Nuclear Station Unit 2 (TMI-2)  
Operating License No. DPR-73  
Docket No. 50-320  
SNM Accountability

Dear Sir:

By NRC letter dated October 17, 1985, GPU Nuclear was granted exemption from certain requirements for periodic inventory and reporting of the special nuclear materials (SNM) balance for Three Mile Island Unit 2 (TMI-2). As a condition of the exemption, GPU Nuclear was required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment was referred to in the exemption as the "post-defueling survey." GPU Nuclear letter 4410-88-L-0162 dated September 30, 1988, submitted the initial Post-Defueling Survey Report (PDSR) sub-reports.

As stated in that submittal, the PDSR documents the GPU Nuclear assessment of the amount of residual SNM in the various facilities, systems, and components of the plant and describes the methodology utilized to determine the quantity of SNM in each case. The attached PDSR (Enclosure 1) transmits the post-defueling survey results for the TMI-2 Reactor Vessel (RV). In addition, Enclosure 2 provides replacements for the "Overview/Summary" (properly titled the "Executive Summary"), "SNM Plan Summary," and "Introduction" sections of the PDSR document. The PDSR document is officially complete.

The RV PDSR sub-report completes the final assessment of the quantity of residual SNM at TMI-2 for accountability purposes. A total SNM accounting is ongoing; submittal of the Material Balance Report (DOE/NRC Form 742) is scheduled for April 30, 1993.

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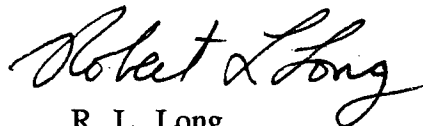
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Page 2

The exemption from the annual physical inventory requirements (10 CFR 70.51(d)) granted to GPU Nuclear in the above referenced NRC letter cites no condition or date of expiration of that exemption. Accordingly, since the residual inventory is fixed, as reported in the PDSR, GPU Nuclear concludes that the exemption from 10 CFR 70.51(d) continues in force. Please advise us if that interpretation is incorrect.

Sincerely,



R. L. Long  
Director, Corporate Services/TMI-2

EDS/dlb

Attachment

cc: T. T. Martin - Regional Administrator, Region I  
M. T. Masnik - Project Manager, PDNP Directorate  
L. H. Thonus - Project Manager, TMI  
F. I. Young - Senior Resident Inspector, TMI

REACTOR VESSEL

**ENCLOSURE 1**

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

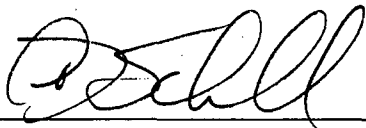
THE REACTOR VESSEL

Approved: JJ Bynum 1/18/93  
Manager, TMI-2 Engineering

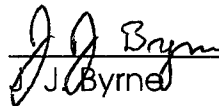
Approved: Robert E. Ryan 1/21/93  
TMI Licensing Director

TMI-2 INTERNAL REVIEWS


Author:

 1/18/93  
E. D. Schrull

Responsible Technical Reviewer:

 1/18/93  
J. J. Byrne

PDSR Committee Chairman:

 1/21/93  
R. E. Rogan



## SUMMARY

The estimate of record of the amount of uranium dioxide ( $\text{UO}_2$ ) remaining in the Reactor Vessel (RV) is 925 kg with an uncertainty of  $\pm 40\%$  expressed as one-standard deviation. The estimate of record is based on underwater video inspections and passive neutron measurements of the residual fuel adjusted to account for measurement bias. The passive neutron measurement technique utilized the results of the video inspections to divide the RV into nine zones which separated the major fuel deposits by elevation. Water was removed from the flooded RV in stages that corresponded to the zones so that the water could be used as a shadow shield to separate the fuel deposits by zone. This technique simplified analysis and improved the accuracy of the passive neutron measurement.

Because of the technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences, the presence of neutron absorbers and boron-induced ( $\alpha, n$ ) reactions, a distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel. The RV estimate of record was determined by adjusting the passive neutron measurement estimate to consider the measurement biases identified by the Rasmussen Committee.

The total amount of residual  $\text{UO}_2$  (i.e., 925 kg) is about 50% larger than the video estimate. The difference between the earlier video estimate and the estimate of record is attributed to the indeterminate uncertainty associated with the visual estimates that were developed from review of the videotapes collected during the video inspection effort.

# TMI-2 POST-DEFUELING SURVEY REPORT

FOR

THE REACTOR VESSEL

## 1.0 INTRODUCTION

This report presents the analysis of the Three Mile Island Unit 2 (TMI-2) Reactor Vessel (RV) uranium dioxide inventory. It is the final report in a series of reports generated to fulfill the requirements of the TMI-2 SNM Accountability Program. Unless noted, all statistical data uncertainties are expressed in terms of one standard deviation (one sigma or  $1\sigma$ ).

A video inspection of the flooded Reactor Vessel (RV) internals was completed in early 1990. The results of that survey and others were presented in the Defueling Completion Report (DCR) (Reference 1) which was supplemented following the NRC-sponsored Post-Lower Head Sampling Program (Reference 1a). While the fuel estimates were acceptable for the purposes of the DCR, the video inspection of the RV was not considered to be of sufficient quality for final Special Nuclear Material (SNM) accountability. For this reason, passive neutron measurements were performed to provide the final estimate of the quantity of residual fuel within the RV as well as to estimate the uncertainty. Because of the technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences,

the presence of neutron absorbers and boron-induced ( $\alpha,n$ ) reactions, a distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel.

This document outlines the results and uncertainties, as well as the specific measurement and analytical tools used to achieve the estimate of record of the residual fuel in the RV. In this report, residual fuel refers to  $UO_2$  and fuel debris refers to a composite material that was the product of the accident.

Section 2, "Background," describes the accident scenario and subsequent cleanup activities. Also discussed are the location and condition of fuel that was relocated during the accident.

Section 3, "Methods," describes the video analysis, the passive neutron measurement technique and equipment used, and the results of the Rasmussen Committee's review that was incorporated in the estimate of record of residual fuel in the TMI-2 RV.

Section 4, "Analysis," explains in detail how the estimate of record of fuel in the RV was calculated and discusses the supporting data, assumptions made, and the assigned uncertainties.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of residual  $\text{UO}_2$  remaining in the RV and supporting rationale leading to a conclusion that the estimate is reasonable based upon the available data and analysis performed.

## 2.0 BACKGROUND

During the March 1979 accident, the TMI-2 nuclear fuel was overheated due to a loss of coolant. Core temperatures were sufficient to remove the cladding by melting or gross oxidation. This process advanced for about 174 minutes after initial loss of coolant. At this time, the "2B" reactor coolant pump (RCP) operated for a few minutes transporting cold water to the core causing further thermal-mechanical degradation of the cladding and pellets. About 30 minutes later, the Emergency Core Coolant System was used to refill the RV, Reference 1.

At about 225 minutes, instruments indicated, and later analysis revealed, a major core relocation. Visual inspections conducted during defueling indicated a flow of molten core material to the Lower Core Support Assembly (LCSA). Approximately 16 hours after the start of the accident, the RCP "1A" was restarted effectively ending the thermal degradation process.

One effect of the accident was to shatter the ceramic uranium dioxide pellets,  $\text{UO}_2$ , and to mix  $\text{UO}_2$  with other metal oxides creating a gravel-like debris bed covering a mass of

resolidified composite material. Subsequent defueling operations shattered the resolidified material into gravel-like material visually indistinguishable from the other debris.

After defueling, the residual fuel was observed to be composed of finely divided material, thin crusts, and solid masses. Small amounts of debris were left in hard to access locations, and a light "dusting" of residual fines was widely deposited on horizontal surfaces after the last underwater vacuum cleaning operation. Crusty resolidified material was distributed along surfaces that tended to channel the molten fuel. Solid masses of the same material were observed in a few LCSA peripheral coolant flow holes that could not be defueled. The  $UO_2$  content of the fines and crusty material is similar; with a larger bulk density for the crusty material.

### 3.0 METHOD

#### 3.1 VISUAL EXAMINATION METHOD

A license condition for transition from Facility Mode 1 (i.e., active defueling) to Facility Modes 2 and 3 (i.e., final preparation for PDMS) was confirmation that the RV had been defueled to the extent reasonably achievable. Video examinations of the RV performed during and subsequent to the completion of defueling were used to quantify the amount of fuel left in the TMI-2 RV. The

results of those visual examinations were presented in References 1 and 1a.

Approximately 133,000 kg of core debris, including approximately 94,000 kg UO<sub>2</sub>, remained in the RV after the accident. Extensive defueling operations were performed in the RV to remove core debris resulting from the accident. The determination of residual quantities of core debris and its location in the RV was an ongoing process from the start of defueling. Techniques included video inspection and sample acquisition and analysis.

Extensive visual examination of nearly all accessible parts of the RV internals were made during and following RV defueling. The resulting videotapes were analyzed to infer the amount, form, and location of the residual fuel in the RV (References 1 and 1a). The physical extent of debris deposits was mapped in three dimensions, using known reference points or landmarks as dimensional indicators. Given good lighting conditions, the vertical and lateral extent could be estimated fairly accurately, but depth (i.e., dimension along the line of sight) was much less easily determined. The physical distribution was then used to estimate volume. Surface texture and other subtle factors were used to identify and compare the subject debris deposit to other similar material for which sample analysis data existed. The sample data for fuel material was then used to estimate the density and composition of the deposit.

The engineers who worked on the project developed a conservative, yet realistic, estimate of the quantity of residual fuel. For instance, if a flow hole in an LCSA plate could not be seen by the video cameras, the engineer was to assume that the hole was completely full of fuel debris. For this reason, when the visual residual fuel value was reported, it was assumed to represent a conservative upper bound of the quantity of residual fuel in the TMI-2 RV. However, the visual estimate was not presented as the final residual fuel value for the TMI-2 RV due to its indeterminate range of uncertainty. Rather, passive neutron measurements were conducted as part of the SNM Accountability Program.

### 3.2 PASSIVE NEUTRON MEASUREMENT TECHNIQUE

Passive neutron measurements were performed to provide the final estimate of the quantity of residual fuel within the TMI-2 RV. The passive method equates the measured number of neutrons released by the core debris to the remaining fuel quantity. Passive neutrons are generated by spontaneous fission of the fuel debris and by  $(\alpha, n)$  reactions with oxygen and boron. The alpha particles are largely produced by the decay of the transuranics.

It was postulated that the video estimates could be used to predict neutron yields for the visualized deposits. Residual fuel quantities could then be assessed by comparing neutron measurements with predictions. For simple systems, the



product of the measured neutron/second (n/s) and the predicted kg UO<sub>2</sub> per n/s would provide the fuel quantity of record. Unfortunately this simple analysis was complicated by geometry, and by the resolution of the video inspection method (References 2 and 2a).

The video-estimated fuel deposits were not uniform. Correlating deposits with locations revealed that the thickness of stainless steel shielding the fuel was also variable. Partially known, non-uniform deposits behind absorbers of varying thickness provide a complex fuel/source distribution and neutron absorption quality. Therefore, it was determined that a single measurement made near the center of the dry RV would not be useful to resolve the complex source term for an appropriate fuel estimate.

The visually-determined fuel distribution was used to divide the vessel into regions that could be usefully separated by water level. The implementation of a controlled draindown divided the RV internals into nine vertical zones as shown in Figure 1. Zone elevations are given in Table 1. Multiple passive neutron measurements, assisted by the RV water shield, were used to reduce the complexity for a better fuel assay. The water shield was created by draining the RV to successive zone elevations.

The distances between the entryway into the RV, the top of the chimney, and various zones were based on physical measurements and engineering drawings of the pressure vessel and vessel internal components.

### 3.3 PASSIVE NEUTRON MEASUREMENT BIASES

The passive neutron measurement technique is subject to systematic biases that could significantly affect the resultant estimate of record. The technical complexities involved in the passive neutron measurement program, including complex geometries, structural interferences, the presence of neutron absorbers and boron-induced ( $\alpha,n$ ) reactions, make these systematic biases difficult to quantify with any reasonable accuracy.

A distinguished group of scientists, chaired by Dr. N. Rasmussen, Massachusetts Institute of Technology, and including Dr. H. Menlove, Los Alamos National Laboratory, and Dr. G. Knoll, University of Michigan, was commissioned to conduct an independent review of the measurement data and the estimate of residual fuel. The Rasmussen Committee concluded that measurement biases were present in the passive neutron measurement technique (Reference 3). These biases and their impact on the passive neutron measurement estimate are discussed in Section 4.3.

### 3.4 HARDWARE FOR THE PASSIVE NEUTRON MEASUREMENT

Four assemblies comprise the hardware used for the passive neutron measurements (Reference 2). They are the detector and stand assembly; the contamination control box; the calibration source deployment system; and the RV liquid level monitor.

#### 3.4.1 DETECTOR AND STAND ASSEMBLY

The detector assembly consisted of three He-4 detectors and coupled preamplifiers that were shielded by about 15 cm of lead and steel. The shield was used to reduce the radiation level near the detectors to about 0.5 R/h. The assembly was designed to be deployed by the RB polar crane using a 50' long steel chain. A special signal/power cable bundle was fabricated to interconnect the detector's preamplifiers to their amplifiers, power supplies, and a computer.

A stand assembly was designed to support the 2 ton detector assembly. The stand was located on the 347' El. between the tool decon facility and the "B" D-ring shield wall. Close proximity and clear visual space between the stand and associated electronics located in the contamination control box aided adjustments.

### 3.4.2 CONTAMINATION CONTROL BOX

A housing was provided for a computer, a NIM-bin containing several modules, an RS232 modem, and an electrical cable terminator plate. The housing was defined as the contamination control box (CCB).

The main purpose of the CCB was to shield against electrically-generated noise and to allow flexibility in terminating the various signal/power cables in a contamination-free environment.

### 3.4.3 CALIBRATION SOURCE DEPLOYMENT SYSTEM

For calibration and operational checks, an Am-241-Be neutron source was lowered into the RV through three stainless steel tubes. The tubes extended down from the shielded work platform to the bottom of the RV. The tube used for calibration was located inboard of the Core Support Structure (CSS)-Upper Core Support Assembly (UCSA)-LCSA.

An 1/8" stainless steel ball and bead chain was used to support the source. Incremental chain length distance uncertainties were less than 1/4". The position of the source was monitored by calibrating the ball and bead chain for length. The calibration marks were white balls against a flat

black painted chain. Readout of source position was by a video monitor located in the command center which was connected to a camera attached to the reel assembly. Readout in the RB was not needed as the source controller was located within clear sight of the source reel.

#### 3.4.4 RV LIQUID LEVEL INDICATOR

A staged precision bubbler was used for liquid level readout. The heart of this device was a pressure transducer that was rated for an accuracy of 0.1%. A water depth of about 11.5' was covered by a 5 psig transducer, and the liquid level was followed during pump-out by moving the bubbler elevation in large stages. The error determined by quadrature\*, anywhere in this span, was 0.23 inches.

### 4.0 ANALYSIS

#### 4.1 VISUAL ESTIMATE

The visual estimate of the residual fuel in the TMI-2 RV was 630 kg. The documentation in References 1 and 1a provides a description of the analytical

\* - Square root of the sum of the squares of the individual errors.

results of the video examination of the TMI-2 RV. The data presented showed a residual fuel inventory of 609 kg. However, for comparison with the passive neutron measurement estimate and the estimate of record, the 21 kg extant in the hot and cold leg nozzles (Reference 4) was added to achieve the stated 630 kg (Figure 2).

As shown on the archived videotapes, the residual fuel in the TMI-2 RV consists primarily of finely divided, small particle-size sediment in inaccessible holes, crevices, and corners; surface films; and resolidified material either tightly adherent to the RV components or inaccessible for defueling. Approximately 50 samples of residual core debris were analyzed for density and fuel (i.e.,  $UO_2$ ) content. An average density and representative fuel content was determined for each type of material. A computer model of the RV and its internal components was developed which defined the dimensional characteristics of the remaining RV internals and, more importantly, the spaces where core debris could reside. Then, based on observation, the actual core debris was spatially represented in the computer model.

Computer software was developed which used the information contained in the geometry model to calculate the volume of modelled debris. Other visual factors such as surface texture, shape, apparent hardness or porosity, friability, color, and location were used to categorize the residual core debris into one of the three

types discussed above. A mass estimate was then obtained by multiplying the estimated core debris volume by the average density and UO<sub>2</sub> percentage corresponding to the particular type of material (i.e., resolidified material, loose debris, or surface film).

The benefit of the video assay was not only in identifying where fuel deposits were located but also in identifying where no fuel deposit greater than the video camera resolution of about 1 mm existed. However, the video assay had its limitations, such as, lack of access to all areas of the RV, translation of a two-dimensional image into a three-dimensional portrait, and the resolution of the video cameras. For these reasons, no uncertainty was ascribed to the video estimate and, more importantly, the video estimate was not considered sufficiently accurate to be used as the sole basis for the estimate of record for SNM accountability purposes. The SNM Accountability Program required a more technically exact measurement technique; thus, the passive neutron measurement technique was developed.

#### 4.2 PASSIVE NEUTRON MEASUREMENT ANALYSIS

The passive neutron measurements produced raw neutron count rates for a series of locations within the RV under known conditions. Fuel increments in each zone were determined by satisfying coupled simultaneous sets of equations that

represented the series of measurements made in each zone (Reference 2). A neutron measurement was made with the water level at the bottom of one zone (e.g., Zone 3) and the detector located within that zone. The water level was then dropped to the bottom of the next zone (e.g., Zone 4) and with the detector in the same location (e.g., within Zone 3), another neutron measurement was made. The detector was then lowered to the next zone (e.g., within Zone 4) and the measurement process was repeated. Neutron attenuation in stainless steel, lead, and water was represented by simple exponentials with increasing thickness. Gamma ray transport codes were used with material and density selected to force the codes to match the empirical attenuation characteristics.

The fuel locations and a portrait of the remaining RV internal structure defined by the video inspection were used to produce minimum and maximum fuel values. The minimum fuel value was attained by adding fuel to areas nearest the detector until the actual measured neutron count rate equalled the computer-modelled count rate. Conversely, the maximum fuel value was determined by adding fuel to areas furthest from the detector until the same match was achieved. The average of the minimum and maximum values for each zone was the passive neutron measurement estimate. The passive neutron measurement estimate of residual fuel in the TMI-2 RV was 1322 kg.



### 4.3 EVALUATION OF MEASUREMENT BIASES

Since the passive neutron estimate was more than twice the visual estimate, which was considered conservative, the Rasmussen Committee was commissioned to evaluate the measurement data, the passive neutron measurement technique, and residual fuel estimate.

The Rasmussen Committee concluded that there were five systematic measurement biases in the passive neutron measurement technique and analysis. These biases, related to boron variations,  $\text{UO}_2$  particle size, calibration error, data analysis, and in-scattering of neutrons, were difficult to quantify with any reasonable accuracy.

#### 4.3.1 BORON VARIATIONS

Neutron yield from the TMI-2 core debris is primarily a function of three distinct components: spontaneous fission of transuranics,  $(\alpha, n)$  reactions from oxygen, and  $(\alpha, n)$  reactions from boron. The dominant neutron source term is  $(\alpha, n)$  reactions from boron; the neutron yield from this component is proportional to the thickness of the boron layer on the surface of the  $\text{UO}_2$ . The Rasmussen Committee concluded that a bias in the passive neutron measurement was the inability to fully drain the borated water without a buildup of boron on the residual fuel. This

resulted in an increase in the neutron yield from residual fuel in Zones 1 through 5. The Rasmussen Committee ascribed an estimated bias of +15% for this effect in Zones 1 through 5.

#### 4.3.2 UO<sub>2</sub> PARTICLE SIZE

Fuel particle size has a strong affect on neutron yield. A UO<sub>2</sub> pellet will yield orders of magnitude less neutrons per gram than UO<sub>2</sub> powder with a surface coating of boron. Further, small dust-size particles would adhere to all surfaces; whereas, larger particles would tend to drop to the bottom of the vessel. Therefore, the Rasmussen Committee concluded that the passive neutron measurements were biased because the neutron yield calibrations used TMI-2 fuel samples that were comprised of a non-representative lesser quantity of fine particles. Defueling videos provided extensive observations of fine powder deposits in Zones 1 through 5. Therefore, the Rasmussen Committee ascribed a total estimated bias of +45% from these effects to Zones 1 through 5.

#### 4.3.3 CALIBRATION ERROR

A measurement bias was introduced by using an AmBe source for the primary calibrations. The calibration technique caused the detector to be more efficient for the larger-size particles extant in the bottom regions of

the RV. An additional source of calibration error was the variation in angular distribution of the neutrons from the AmBe source as compared to the UO<sub>2</sub> sample. The bias for these effects was estimated at -5% on a total vessel basis.

#### 4.3.4 DATA ANALYSIS

Nine grab samples of fuel debris were used for neutron calibration. The results from one of the samples was omitted from the calculation of neutron yield. Including this sample in the average would increase the neutron yield by approximately 5%. Therefore, the bias for this effect was estimated at +5% on a total vessel basis.

#### 4.3.5 NEUTRON IN-SCATTERING

The passive neutron measurement technique corrected the data for neutron absorption and out-scatter, but no credit was given for neutron in-scatter. A bias was introduced through the use of gamma-ray computer codes that do not account for neutron scatter in non-line-of-sight structural steel. The Rasmussen Committee ascribed an estimated bias of +20% on a total vessel basis for this effect.

#### 4.4 ESTIMATE OF RECORD

Since the residual fuel result of the passive neutron measurements was so much larger than the visual estimate and considering the Rasmussen Committee's discussion of systematic biases, it was decided that the estimate of record of the residual fuel in the TMI-2 RV for SNM accountability purposes would be determined by adjusting the results of the passive neutron measurements to account for the biases discussed in Section 4.3. The biases were assigned to specific RV zones and then summed per zone (Table 2). The passive neutron measurement estimate was adjusted accordingly resulting in a zonal estimate of record (Table 3). The estimate of record of the residual fuel in the TMI-2 RV is 925 kg. Figure 2 provides a comparison of the three fuel estimates, i.e., visual estimate, passive neutron estimate and estimate of record, for the TMI-2 RV on a per zone basis.

#### 4.5 ANALYSIS OF UNCERTAINTY

The passive neutron measurement report (Reference 2) includes an uncertainty estimate of approximately  $\pm 15\%$  for the estimated non-bias adjusted RV content of 1322 kg. The Rasmussen Committee judged the uncertainty to be  $\pm 40\%$  when bias adjustments are made to the passive neutron estimate. This uncertainty was based on the following assumptions:

1. Approximately two-thirds of the neutrons counted are not related to the amount of uranium present;
2. The fuel source material is in a wide variety of forms; each with a different neutron yield per gram of uranium;
3. The highly asymmetrical location of the steel complicates evaluation of the in-scattering effect; and
4. The distribution of boron changes during water removal.

It is noteworthy that if two-sigma uncertainties (i.e., 95 % confidence level) are considered, the estimate of record (925 kg) with its estimated two-sigma uncertainty of 80% certainly bounds the visual estimate (630 kg) plus any reasonable associated uncertainty, and essentially bounds the unadjusted passive neutron estimate of 1322 kg plus its two-sigma uncertainty (30%). Figure 3 provides a visual representation of the uncertainty evaluation.

Therefore, it appears reasonable to adopt the Rasmussen Committee's assignment of an uncertainty of  $\pm 40\%$  to the TMI-2 RV estimate of record.

## 5.0 CONCLUSION

The estimate of record of the amount of uranium dioxide ( $\text{UO}_2$ ) remaining in the TMI-2 RV is 925 kg. This estimate has an associated uncertainty of  $\pm 40\%$ . Since the RV has

been drained to the extent that less than 10 gallons of water remain, no fuel movement into or out of the RV is anticipated, and this quantity of residual fuel is considered fixed.

This estimate of record is based on a detailed, non-destructive visual examination and physical measurement of the RV residual fuel that was adjusted in accordance with the recommendations of a panel of distinguished scientists and evaluated by a group of independent external reviewers. It is understood that the bias adjustments are somewhat judgemental due to the difficulty experienced in their quantification. However, the reviewers were in general agreement regarding both the estimate of record and its associated uncertainty. Considering the information provided by the visual estimates and the circumstances under which the passive neutron measurements were made, the estimate of record of 925 kg represents a realistic estimate of the quantity of residual fuel extant in the TMI-2 Reactor Vessel and the consensus of the reviewers commissioned to assess the results for the TMI-2 SNM accountability project.

## REFERENCES

1. GPU Nuclear letter 4410-90-L-0012, "Defueling Completion Report, Final Submittal," dated February 22, 1990.
- 1a. GPU Nuclear letter 4410-90-L-0026, "Results Of Post-Lower Head Sampling Program Cleanup," dated April 12, 1990.
2. GPU Nuclear memorandum, 4240-92-114, "Results of Passive Neutron Measurement Program," dated August 21, 1992.
- 2a. Letter, C. Distenfeld to R. Rogan, dated September 18, 1992, transmitting the report, "TMI-2 Reactor Vessel Fuel Assay."
3. GPU Nuclear memorandum, 5400-92-0045, "Final Transmittal of Rasmussen Review Committee Reports," dated August 28, 1992.
4. GPU Nuclear memorandum, 4850-90-008, "Fuel Assessment Of Hot, Cold, and Core Flooding Nozzles, Tool, Tool Rack, and Tool Decon Facility," dated February 12, 1990.
5. GPU Nuclear letter, C312-92-2080, "TMI-2 Reactor Vessel Criticality Safety Analysis," dated December 18, 1992.

TABLE 1

ZONE ELEVATIONS

<u>ZONE</u>	<u>ELEVATION</u>	<u>DEPTH</u>
0, top of chimney	337.43'	00.00'
1, top of flange	322.48'	14.95'
2, top of hot leg	317.00'	20.43'
3, bottom of cold leg nozzles	313.96'	23.47'
4, bottom of CSS	312.06'	25.36'
5, mid-plane of UCSA	305.17'	32.25'
6, bottom of USCA	298.29'	39.14'
7, top of forging	296.91'	40.52'
8, top of IGSP	295.74'	41.69'
9, empty	291.07'	46.36'



TABLE 2

CORRECTION FACTORS FOR MEASUREMENT BIAS

<u>CORRECTION FACTOR</u>	<u>BIAS</u>	<u>APPLICABLE ZONES</u>
Boron Variations	+15 %	1 - 5
UO <sub>2</sub> Particle Size	+45 %	1 - 5
Calibration Error	- 5 %	1 - 9
Data Analysis	+5 %	1 - 9
In-Scattering +20 %	1 - 9	

SUMMATION OF BIASES

Zones 1 - 5	Bias = 80 %
Zones 6 - 9	Bias = 20 %

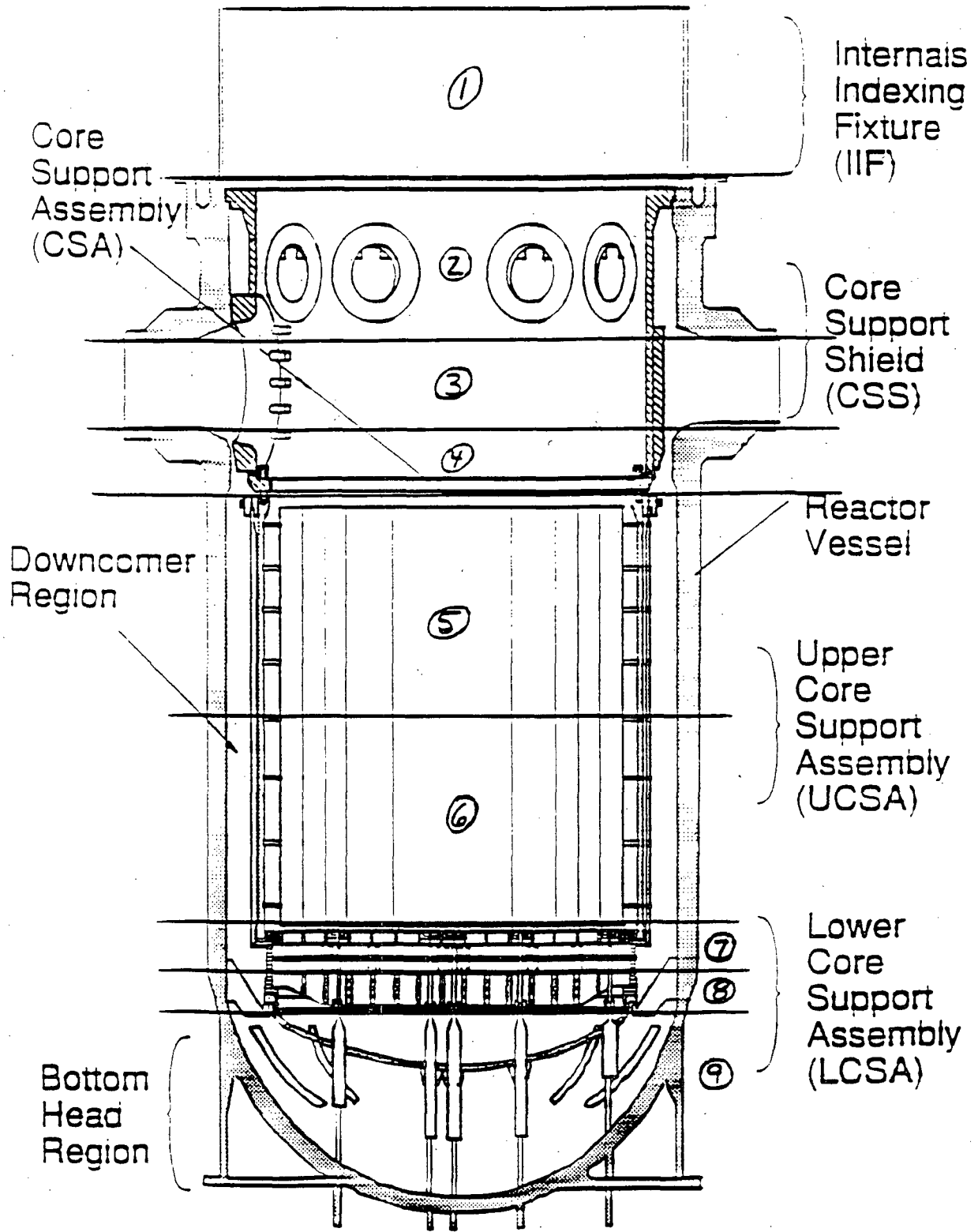
TABLE 3

REACTOR VESSEL CALCULATION OF THE ESTIMATE OF RECORD

<u>ZONE</u>	<u>PASSIVE NEUTRON MEASUREMENT ESTIMATE</u>	<u>CORRECTION FACTOR</u>	<u>ESTIMATE OF RECORD</u>
1	10	1.8	6
2	225	1.8	125
3	150	1.8	83
4	99	1.8	55
5	154	1.8	86
6	387	1.2	323
7	113	1.2	94
8	89	1.2	74
9	95	1.2	79
TOTALS	1322		925

FIGURE 1

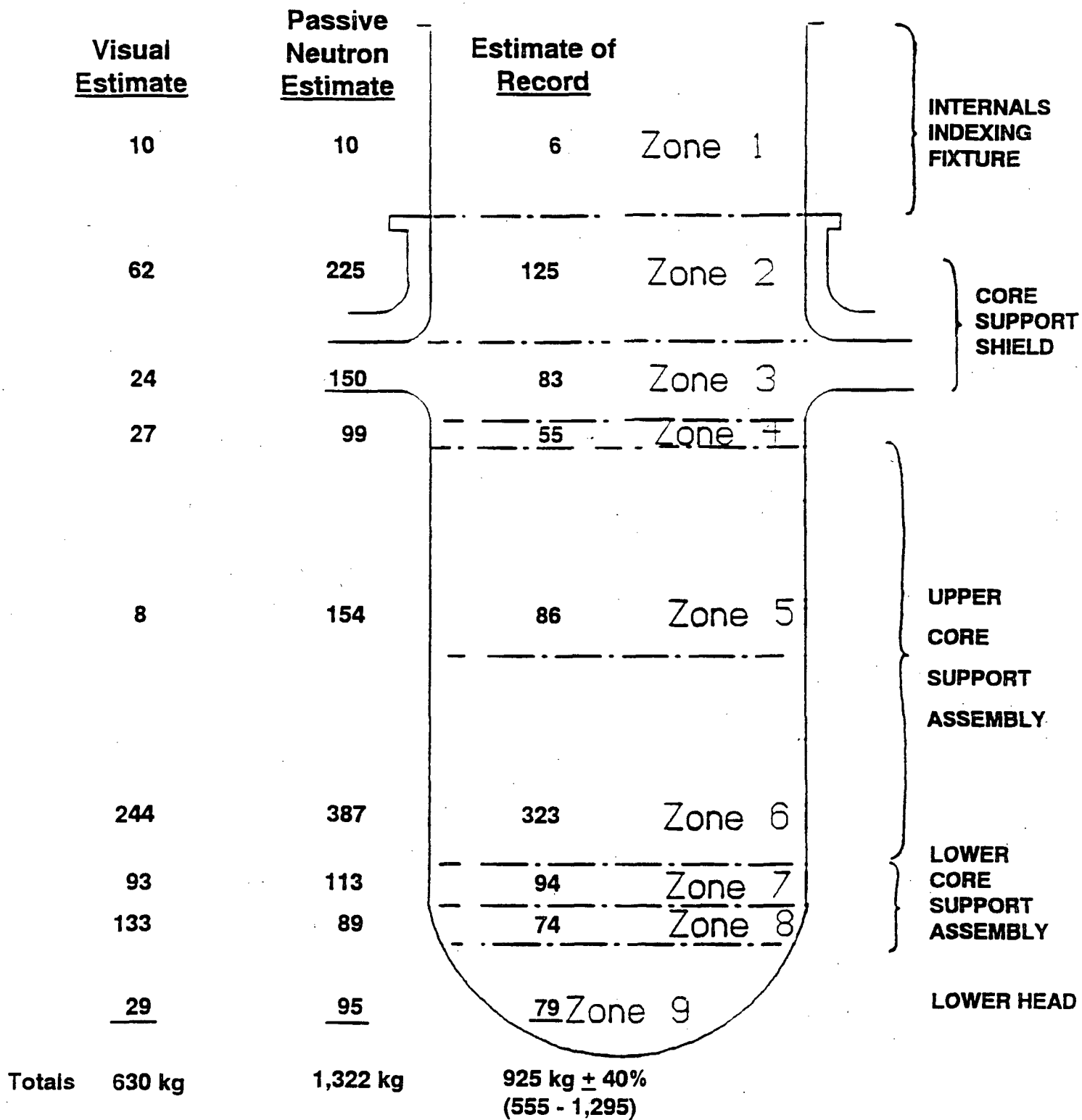
RV Vertical Zones



Reactor Vessel Cutaway View

FIGURE 2

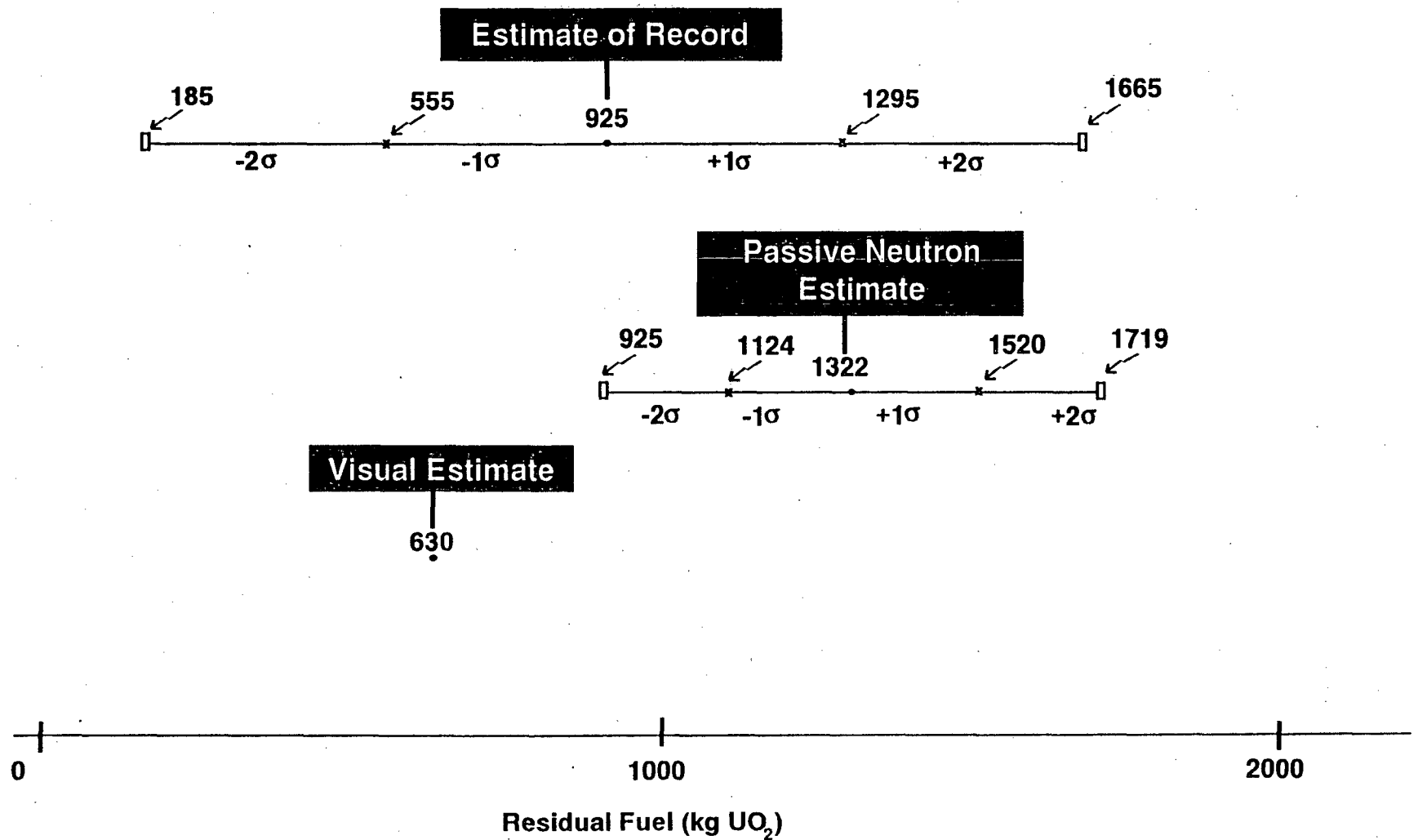
**Comparison of RV Fuel Estimates**



**Estimate of Record = 925 kg**

# FIGURE 3

## Evaluation of Estimate of Record Uncertainty





ENCLOSURE 2

## EXECUTIVE SUMMARY

The initial Three Mile Island Unit 2 (TMI-2) core contained 94.0 tonnes (94,000 kg) of uranium dioxide (UO<sub>2</sub>). Three months of full power operations, defueling, and fuel shipment to the Idaho National Engineering Laboratory reduced the inventory to approximately 1100 kg UO<sub>2</sub><sup>1</sup> or about 1.2% of the original core loading. The purpose of this document is to discuss the process used to determine the total amount of UO<sub>2</sub> remaining in the TMI-2 facility, and to document the final fuel estimate of record.

Table 1 depicts the final residual fuel distribution of the TMI-2 facility. The total quantity of residual fuel was approximately 1100 kg with an uncertainty of about  $\pm 370$  kg. The uncertainty (one standard deviation -  $1\sigma$ ) was calculated as the square root of the sum of squares of the individual uncertainties. In general, the individual uncertainties were the standard deviations of the "estimates of record." For the Auxiliary and Fuel Handling Buildings (AFHB), Reactor Building Miscellaneous Components, Plenum, and the Reactor Vessel (RV) Head, the uncertainties were reported as asymmetric quantities. For these cases, the averages of the upper and lower error bounds were taken as the standard deviation. The uncertainty in the residual fuel in the RV dominated the total uncertainty. It is useful to note that the total quantity of residual fuel outside of the RV was about 170 kg; an amount less than one-half of the standard deviation of the residual fuel in the RV.

The collection of PDSR sub-reports provide a comprehensive and systematic accounting of all

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<sup>1</sup> Given an average fuel enrichment of 2.24 wt % U-235, the total quantity of contained U-235 at TMI-2 is less than 22 kg.



residual fuel in the TMI-2 facility. The results of each survey of an area, system, or component containing special nuclear material (SNM) are detailed in a separate PDSR sub-report and a compilation of these results forms the basis for the final SNM accounting for TMI-2.

To facilitate the review and comprehension of this compilation of sub-reports, a section describing the SNM Program Plan and the specific requirements and considerations for measurements, quality assurance, and program management is included. The Introduction section describes the format of the PDSR sub-reports, provides a graphic illustration of the SNM areas, systems, and components being surveyed, and explains the various methods and techniques used to measure SNM and analyze the data. Each sub-report provides specific analytical and/or measurement results, the "estimate of record" of the residual fuel for the given area, system, or component, and a discussion of the measurement methods and analytical techniques utilized, as well as the level of uncertainty of the results. Each sub-report received an independent external review by Dr. R. Brodzinski, Battelle Pacific Northwest Laboratories, Dr. F. Tingey, University of Idaho, and either Dr. T. Sampson or Dr. P. Russo, Los Alamos National Laboratory.

The fuel assay process extended over several years. The earlier PDSR sub-reports tended to cover the fuel assay results for specific components (e.g., the RV Plenum). More recent reports served to group together components common to a specific location, such as the AFHB and the Reactor Coolant System. For ease and continuity of discussion, the detailed data, measurement information, calculations, and analysis were not included in the sub-reports. The individually referenced support material was archived for retrieval by interested parties.

TABLE 1

POST-DEFUELING DISTRIBUTION OF RESIDUAL UO<sub>2</sub>  
IN THE TMI-2 FACILITY

PDSR SECTION TITLE	UO <sub>2</sub> kg	UNCERTAINTY kg
Plenum	2.1	± 1.9
Letdown Coolers	3.7	± 2.0
Pressurizer	0.3	± 0.2
Reactor Vessel Head	1.3	± 0.9
Reactor Building Basement	1.3	± 0.7
"A" and "B" OTSG	62.3	± 9.7
Auxiliary and Fuel Handling Buildings	11.5	± 5.8
Reactor Building Miscellaneous Components	64.0	± 26.9
Reactor Coolant System	25.8	± 11.1
Reactor Vessel	925	± 370
<b>TMI-2 Facility Total Fuel</b>	<b>1097</b>	<b>± 371</b>

## SUMMARY OF TMI-2 SPECIAL NUCLEAR MATERIAL ACCOUNTABILITY PLAN

### **PURPOSE**

The Special Nuclear Material (SNM) Accountability Plan was initially issued in April 1987. The purpose of the SNM Accountability Plan was to define the method of SNM accountability, the Quality Assurance Program incorporated in the SNM Accountability Program, the areas, systems, and components to undergo formal SNM measurement, and those that did not require SNM assessment. The Plan also described the integration of programmatic ALARA in SNM assessment activities.

As defined in the SNM Accountability Plan, the post-defueling survey was a process by which the entire TMI-2 plant was surveyed to identify areas known to contain, or that could contain, residual SNM and to quantify that SNM. The quantity of SNM in each applicable area was determined. The accomplishment of the required SNM measurements and associated engineering analyses, and a determination of the estimate of record of the total quantity of residual fuel at TMI-2, constitute completion of the post-defueling survey. The SNM Accountability Plan describes the process by which the post-defueling survey was conducted.

## **BACKGROUND**

The March 1979 accident resulted in significant damage to the core and the subsequent release of fuel and fission products into the Reactor Coolant System (RCS) and other closely related systems. The post-accident TMI-2 core consisted of loose fuel pellets, solidified fuel, structural metal components (e.g., end fittings), loose rubble, and partial fuel assemblies. This collection of material is generically referred to as core debris. As a result of the core condition, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

Core debris was removed from the TMI-2 reactor, loaded in special containers, and shipped to the Department of Energy Idaho National Engineering Laboratory (DOE INEL) facility in Idaho. Each shipment was accompanied by a Nuclear Material Transaction Report (DOE/NRC Form 741) which recorded the net weight of the contents of each canister and a best available physical description of the contents. A statement that quantification of the amount of SNM in each canister was not possible also accompanied each shipment as an annotation on the DOE/NRC Form 741.

The canister contents were a mixture of SNM and other core debris and structural materials. There was no feasible method at TMI-2 to determine the exact content of fuel in each canister by isotope with gram specificity. Therefore, SNM accountability for TMI-2 was based on the total measured SNM remaining in the plant after defueling was complete. A final plant

inventory of residual SNM will be reported on the DOE/NRC Material Balance Report (DOE/NRC Form 742).

In October 1985, GPU Nuclear, the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) entered into an agreement (References 1 and 2) that final SNM accountability for TMI-2 would be performed after defueling was completed and would be based upon a thorough post-defueling survey of TMI-2. This post-defueling survey would quantify, as accurately as possible, the amount of residual SNM in plant areas, systems, and components. Implied in this agreement was the understanding that the post-defueling survey would involve all areas, systems, and components where SNM could possibly have been deposited as a result of the 1979 accident and subsequent cleanup activities.

## SUMMARY

Formal SNM assessment activities began in 1987 and continued until 1992. SNM measurements were performed as areas, systems, and components were defueled and/or placed into an isolated configuration that ensured no fuel transport in or out after the SNM survey was completed.<sup>1</sup> The intent of this isolation configuration was to ensure no transport of SNM into or out of an area, system or component previously surveyed which could significantly alter the SNM assessment.

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<sup>1</sup> An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

Some SNM measurements were performed on areas, systems, or components that were not in an isolated configuration. This approach was not preferred and these measurements were utilized for SNM accountability only when analysis demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the measurement. When areas, systems, or components could be physically isolated prior to SNM measurement, the isolation configuration was selected to enhance SNM detection with due regard for system boundaries, piping configuration, and measurement requirements. Following SNM measurements, the configuration was and will continue to be administratively and physically controlled.

A review of the entire TMI-2 facility was conducted to determine where SNM may have been deposited as a result of the 1979 accident and subsequent cleanup activities. Locations were placed into three categories: Category A - locations where SNM is definitely deposited; Category B - locations where it can be reasonably postulated that SNM may be deposited; and Category C - locations where it can be shown that SNM was not deposited. Appendix 1 categorizes each area, system, and component of concern.

All Category A and B areas underwent SNM assessment. Category C areas were identified as not requiring SNM assessment based upon authoritative analysis of the TMI-2 accident (Reference 3) and a review of subsequent cleanup activities.

SNM accountability at TMI-2 was a complex task. Inaccessibility of some areas, systems, and components, high area radiation backgrounds, complex geometries, and the required indirect measurement of fuel complicated physical measurement of SNM quantities. Therefore, several alternative techniques for performing measurement of SNM quantities were used. Reference 5 describes those techniques.

SNM assessment was an Important to Safety (ITS) activity; therefore, the TMI-2 Recovery QA Plan applied to all SNM assessment activities. The SNM Accountability Plan, SNM measurement procedures, and Unit Work Instructions underwent Quality Assurance/Quality Control (QA/QC) review and concurrence. Measurement equipment was maintained and calibrated in accordance with QA/QC requirements. Individual SNM assessment activities included QA/QC verification of essential parameters, as deemed necessary. Records of SNM assessment activities and associated analyses were subject to QA/QC monitoring and auditing. Engineering calculations for SNM assessment were performed in accordance with the TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations."

## **POST-DEFUELING SURVEY REPORTS**

The collection of Post-Defueling Survey Reports (PDSRs) documents the estimates of record of residual SNM in each Category A and B area, system, and component at TMI-2. The quantity of residual SNM was determined through measurements, sampling, and engineering analysis. Each PDSR sub-report contains, as a minimum, a description of the boundaries of the sub-

report, a detailed description of the area, system, or component to be surveyed, a description of its role in the accident and/or subsequent cleanup activities, specific measurement results, an explanation of the measurement method(s) used to obtain fuel measurement data, a discussion of the analysis techniques utilized, the estimate of record of residual fuel, and its associated uncertainty.

#### **DETERMINATION OF SNM IN RADIOACTIVE WASTE AND SAMPLE SHIPMENTS**

The total amount of SNM shipped offsite as radioactive waste and/or as samples was determined in accordance with approved ITS procedures. This amount of SNM shipped was then recorded on the appropriate shipping forms and the DOE/NRC Form 741. The quantities of SNM reported as shipped was included in the calculation which finalized the SNM accountability process at TMI-2.

#### **FINAL SNM ACCOUNTABILITY**

Final SNM accountability for the TMI-2 facility was determined by compiling the results of individual sub-reports (i.e., "estimates of record") of the quantity of residual SNM for each area, system, or component. The amount of fuel shipped to the DOE INEL was determined by subtracting the sum of the final plant inventory and the amount of SNM shipped as radioactive waste from the total plant inventory of SNM as reported on the most recent SNM Material Balance Report (DOE/NRC Form 742) as corrected for decay.



Last reported inventory

- Decay correction
  - Final plant inventory
  - SNM shipped as samples and radwaste
- = SNM shipped to INEL in fuel, filter and knockout canisters

## SNM MEASUREMENTS

SNM measurements were performed on TMI-2 Category A and B areas, systems, and components (Appendix 1). A determination of the residual SNM in each location was based upon individual SNM measurements performed using approved procedures, or by examination and analysis of previously performed fuel characterization measurements. SNM measurements were performed in accordance with an approved procedure or Unit Work Instruction when existing data from previous fuel characterization measurements were insufficient for final SNM accountability. Generic procedures were utilized for measurements which applied standard techniques (e.g., gamma spectroscopy). Unit Work Instructions were utilized for measurements that required special, one-of-a-kind techniques (e.g., OTSG tube film SNM measurement). Data sheets attached to each SNM measurement document were used to record the data required for post measurement analysis.

SNM measurement documents were reviewed and approved in accordance with TMI-2 Unit Procedure 4000-ADM-1218.02, "TMI-2 Document Evaluation, Review and Approval." SNM measurement documents required internal concurrence in accordance with TMI-2 Procedure 4000-PLN-4420.02, "SNM Accountability Plan." SNM measurement documents were archived in CARIRS.

### **SNM MEASUREMENT TECHNIQUES**

Post-defueling SNM assessment at TMI-2 was a complex task. Several different measurement techniques were used. Technique selection for each measurement depended upon the configuration of the area, structure, or component assayed, physical access limitations, area radiation dose rates, and the likely nature of the form of SNM deposits. Extensive use was made of gamma scintillation counting, neutron detectors, video inspections, radiochemical analysis of samples, and alpha detection. Several areas were surveyed utilizing two or more techniques. A description of the most common measurement techniques and measurement technique selection criteria can be found in Reference 4.

### **QUALITY ASSURANCE FOR SNM ACCOUNTABILITY**

The results of the SNM Accountability Program were the basis for final SNM accountability at TMI-2. Therefore, the SNM accountability activities were classified as "Important to Safety" and complied with the TMI-2 Recovery QA Plan.

Final SNM accountability was based on a determination of the quantity of residual SNM remaining in TMI-2 areas, systems, and components after defueling was completed. The determination of the residual SNM quantities was based upon measurements performed utilizing QA/QC approved procedures or upon measurement packages that contained previously performed fuel characterization measurements reviewed and approved by QA/QC on an after-the-fact basis.

SNM measurement activities performed via approved procedures or Unit Work Instructions were reviewed by QA/QC for inclusion of hold/witness points. Specific activities (e.g., critical items) were also identified by the document preparer which required QA/QC verification. Activities requiring QA/QC observation required coordination to ensure that QA/QC support was timely and available for the specific activity.

SNM engineering calculations were performed in accordance with TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations." Data utilized in the engineering calculations were obtained from the completed QA approved procedure or Unit Work Instruction data sheets or from QA reviewed and approved data acquisition measurement packages. Calculations were independently verified in accordance with the referenced procedure. SNM measurement packages that identified residual SNM deposits greater than 1 kilogram were submitted for an independent review.

Computer codes utilized to quantify residual SNM were verified by the Fuel Measurement and Analysis Section by benchmarking with accepted industry codes. Verification was documented. Only verified and approved computer codes were utilized. Approved code versions were controlled to preclude unauthorized modification. Code versions utilized in engineering calculations were specifically identified.

Equipment utilized to quantify residual SNM via a QA-approved procedure or Unit Work Instruction was calibrated and operated in accordance with these procedures. Essential equipment identification information (e.g., type, size, configuration) and performance data (e.g., counts, duration of count, location of detector) were recorded on the data sheets.

#### **CONFIGURATION CONTROL OF AREAS, SYSTEMS, OR COMPONENTS CONTAINING RESIDUAL SNM**

In order to ensure that the SNM measurement process was accurate, controls were established to ensure that SNM was not "double counted." Double counting could have occurred if SNM was relocated out of an area, system, or component that had already been measured into a radioactive waste shipment or an area, system, or component still requiring SNM assessment.

Where possible, areas, systems, and components were physically isolated prior to SNM measurement. However, some SNM measurements were performed on areas, systems, or components that were not in an isolated configuration. In these instances, the measurements

were accompanied by an analysis which demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the measurement.<sup>2</sup>

Administrative controls were utilized, as required or appropriate, to maintain physical isolation of areas, systems, or components that have undergone SNM measurement so that transport of SNM into or out of the area, system, or component was precluded. The type of administrative controls depended upon the nature of the area, system, or component.

Piping systems and components connected to piping systems (e.g., pumps, tanks) were isolated utilizing "red" tags via the TMI-2 Administrative Procedure 4000-ADM-3020.04, "Switching and Tagging Safety." Red tags prohibit the changing of position of a component (e.g., valves, electrical breakers). "Red tagged" components served as isolation barriers intended to prevent the transport of residual SNM into or out of areas, systems, or components. If an isolation barrier was removed (e.g., a valve was opened), the need for a repeat measurement of SNM in the affected piping was evaluated.

Open areas (e.g., reactor building basement floor, auxiliary building sump) were controlled by one of two methods to prevent fuel transport. Areas were maintained in a dry condition so that liquids could not displace residual SNM or, if subjected to liquid flow, the liquid was

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<sup>2</sup> An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

sampled. Sampling of the liquid for the presence of residual SNM was performed in accordance with QA approved procedures.

### **ALARA FOR SNM ACCOUNTABILITY**

The SNM Accountability Program was conducted in accordance with the "As Low As Is Reasonably Achievable" (ALARA) principle for radiation exposure. Conduct of the post-defueling survey, as an integral element of the SNM Accountability Program, resulted in the exposure of personnel to ionizing radiation as SNM measurements were performed in the auxiliary, fuel handling and reactor buildings. Personnel radiation exposures were maintained ALARA by limiting the number of measurements to those essential for SNM accountability and by planning each measurement to minimize personnel exposure.

The number of required SNM measurements was limited by utilizing, when possible, previously performed fuel characterization measurements. In addition, SNM measurements were limited to those areas, systems, or components which conceivably contained fuel.

The radiation exposure received by personnel performing SNM measurements was maintained ALARA by proper planning. Individual measurements were designed to include efficient use of time in radiation areas, incorporate lessons learned on dose minimization from previous measurements, and include the use of remote equipment, if possible.

## REFERENCES

1. DOE letter WWB-100-85, "Accountability for the TMI-2 Core," W. W. Bixby to H. M. Burton, EG&G, dated October 8, 1985.
2. NRC letter, "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d), and 70.53," B. J. Snyder to F. R. Standerfer, GPUNC, dated October 17, 1985.
3. NSAC 80-1, "Analysis of Three Mile Island - Unit 2 Accident; Rogovin Report: Three Mile Island, A Report to the Commissioners and the Public."
4. TPO-TMI-187, "Instrument Selection for Residual Fuel Measurements," Rev. 0, dated January 1987.
5. TMI-2 Engineering Procedure 4000-ENG-7310.02, "Engineering Calculations."
6. TMI-2 Administrative Procedure 4000-ADM-3020.04, "Switching and Tagging Safety."
7. TMI-2 Administrative Procedure 4000-ADM-1218.02, "TMI-2 Document Evaluation, Review and Approval."
8. TMI-2 Unit Policy/Plan Procedure 4000-PLN-4420.02, "SNM Accountability Plan."

## APPENDIX 1

### AREA CLASSIFICATIONS

AREA	DESCRIPTION	SNM CATEGORY
AX001	RB Emergency Booster Pump	C
AX002	Access Corridor (Drains)	C
AX003	Access Area (Drains)	C
AX004	Seal Injection Valve Room	B
AX005	MU Pump - 1C	A
AX006	MU Pump - 1B	A
AX007	MU Pump - 1A	A
AX008	Spent Resin Storage Tank - 1B	B
AX009	Spent Resin Storage Tank - 1A	B
AX010	Spent Resin Storage Tank Pump	B
AX011	Aux. Sump Pump Valve Room	A
AX012	Aux. Bldg. Sump Tank Room	A
AX013	Evap. Cond. Tanks, Pumps	C
AX014	RC Evaporator Room	C
AX015a	Cleanup Filters Room	C
AX015b	Cleanup Filter After Room	C
AX016	Cleanup Demineralizer - 2A	C
AX017	Cleanup Demineralizer - 2B	C
AX018	Waste Transfer Pump Room	C
AX019	Waste Disposal Liquid Valves	A
AX020	RC Bleed Tanks - 1B, 1C	A
AX021	RC Bleed Tank - 1A	A
AX022	North Stairwell	C
AX023	Elevator Shaft	C



**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
AX024	Aux. Bldg. Sump Filters	B
AX026	Seal Injection Filters (MU-F4A, 4B)	B
AX027	South Stairwell	C
AX101	Radwaste Disposal Panel	C
AX102	RB Sump Pump Filters	B
AX103	MCC 2-11EB	C
AX104	MCC 2-21EB	C
AX105	Substation 2-11E	C
AX106	Substation 2-21E	C
AX107	MCC 2-11EA	C
AX108	MCC 2-21EA	C
AX109	Nuc. Services Coolers and Pumps	C
AX110	Intermediate Coolers	C
AX111	Intermediate Cooling Pumps and Filters	C
AX112	Seal Return Coolers & Filters (MU-F-3)	B
AX113	Waste Gas Analyzer	C
AX114	MU&P Demineralizer - 1A	A
AX115	MU&P Demineralizer - 1B	A
AX116	MU Tank	A
AX117	MU&P Filters (MU-F2A&B and MU-F5A&B)	A
AX118	Spent Fuel Coolers	C
AX119	Spent Fuel Demineralizers	C
AX120	Spent Fuel Filters	C
AX121	Elevator Shaft	C
AX122	North Stairwell	C

**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
AX123	Access Area	C
AX124	Concent. Liquid Waste Pump	B
AX125	Waste Gas Decay Tank - 1B	C
AX126	Waste Gas Filter Room	C
AX127	Waste Gas Decay Tank - 1A	C
AX128	Valve and Instrument Room	C
AX129	Deborating Demineralizer - 1B	B
AX130	Deborating Demineralizer - 1A	B
AX131	Misc. Waste Tank (WDL-T2)	B
AX132	Corridor between Unit 1 and Unit 2	C
AX133	South Stairwell	C
AX134	Misc. Waste Tank Pumps	B
AX135	Radwaste Disposal Control Panel	C
AX201	North Stairwell	C
AX202	Elevator Shaft	C
AX203	4160V Switchgear - 2-1E	C
AX204	4160V Switchgear - 2-2E	C
AX205	RB Purge Air Supply and Hy. Crot. Exh.	C
AX206	RB Purge Exhaust Unit B	C
AX207	RB Purge Exhaust Unit A	C
AX208	Aux. Bldg. Exhaust Unit B	C
AX209	Aux. Bldg. Exhaust Unit A	C
AX210	FH Bldg. Exhaust Unit B	C
AX211	FH Bldg. Exhaust Unit A	C
AX212	Decay Heat Surge Tank and Substation	C

**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
AX213	Unit Substations and Access Area	C
AX214	Decon Facility	C
AX215	FH Bldg. Supply Unit	C
AX216	Aux. Bldg. Supply Unit	C
AX217	Access Area	C
AX218	Concent. Waste Storage Tank Room	B
AX219	Inst. Racks & Atmosph. Monitor	C
AX220	Caustic Liquids Mixing Area	C
AX221	Caustic Liquids Mixing Area Corr.	C
AX222	South Stairwell	C
AX223	Air Handling Units General Area	C
AX301	Elevator Shaft and Elevator Machine Rm.	C
AX302	North Stairwell	C
AX303	Elevator and Stairwell Access	C
AX401	Roof	C
AX402	Cooling Water Surge Tanks	C
AX403	Damper Room	C
AX501	RB Spray Pump - 1A	C
AX502	RB Spray Pump - 1B	C
AX503	DH Remov. Cooler and Pump - 1A	C
AX504	DH Remov. Cooler and Pump - 1B	C

**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
FH001	Makeup Suction Valve Room	A
FH002	Access Corridor	C
FH003a	Makeup Discharge Valve Room	A
FH003b	Makeup Discharge Valve Room	A
FH004	Westinghouse Valve Room (Mini-Decay Heat)	C
FH005	Mini-Decay Heat Vault	C
FH006	Decay Heat Service Coolers	C
FH007	Neutral & Reclaimed Boric Acid	C
FH008	Neutralizer Tank Pump Room	C
FH009	Neutralizer Tank Room	C
FH010	Reclaimed Boric Acid Tank	C
FH011	Reclaimed Boric Acid Pump	C
FH012	Neutralizer - Tank Filters	C
FH013	Oil Drum Storage Area	C
FH014	Annulus	B
FH101	MU&P Valve Room	A
FH102	East Corridor	C
FH103	Sample Room	C
FH104	West Corridor	C
FH105	Model Room A	C
FH106	Monitor Tanks & Sample Sink Area	B
FH107	Trash Compactor Area	C
FH108	Truck Bay	C
FH109	Spent Fuel Pool A	B

**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
FH110	SDS Spent Fuel Pool	B
FH111	Fuel Cask Storage	B
FH112	Annulus	B
FH201	East Corridor	C
FH202	West Corridor	C
FH203	Surge Tank Area	C
FH204	SPC Area	C
FH205	Annulus	C
FH301	Upper Spent Fuel Pool Area	C
FH302	SDS Operating Area	B
FH303	Upper SPC Area	C
FH304	Annulus	C
FH305	Spent Fuel Pool Access	C

**APPENDIX 1 (cont'd)**  
**AREA CLASSIFICATIONS**

<b>AREA</b>	<b>DESCRIPTION</b>	<b>SNM CATEGORY</b>
RB01	Letdown Coolers Cubicle	A
RB02	Reactor Building Sump	A
RB03	Reactor Coolant Drain Tank Cubicle	A
RB04	Reactor Building Basement (Floor)	A
RB05	Under Reactor Vessel	A
RB06	Letdown Line	B
RB11	Decay Heat Drop Line	A
RB21	Reactor Coolant Pumps	B
RB22	Horizontal RCS Piping	A
RB23	Reactor Vessel	A
RB31	Pressurizer	A
RB32	Steam Generators	A
	Upper Tube Sheet and Tube Blockages	A
	OTSG Tube Surfaces	A
	OTSG Lower Head/J-Legs	A
RB33	Core Flood Tanks - A&B and Drain Lines	A
RB34	Incore Guide Tubes	B
RB35	Plenum	A
RB36	Reactor Vessel Head	A
RB37	Reactor Coolant Hot Legs	A
RB38	Pressurizer Surge Line	A
RB39	Pressurizer Spray Line	B
RB40	Fuel Transfer Canal	B

# POST-DEFUELING SURVEY REPORTS

## INTRODUCTION

### **BACKGROUND**

On October 17, 1985, the Nuclear Regulatory Commission granted GPU Nuclear Three Mile Island Unit 2 (TMI-2) an exemption from certain requirements for periodic inventory and reporting of special nuclear material (SNM) balance (Reference 1). As a condition of the exemption, GPU Nuclear was required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment is referred to in the exemption as the "post-defueling survey."

Post-defueling surveys were performed by conducting measurements for residual fuel in those areas, systems, and components in which residual SNM was assumed to exist and for which no further defueling was planned. A compilation of these survey report packages, entitled "Post-Defueling Survey Reports" (PDSRs), documented the estimate of record of the amount of residual SNM and the methodology utilized to determine the quantity of SNM in each case. PDSR sub-reports have been prepared and submitted for all areas, systems, and components listed in Attachment 1. Attachment 2 graphically illustrates the areas, structures, and components covered by the PDSR sub-reports.

The survey results documented in the PDSR sub-reports were not intended to demonstrate the overall adequacy of TMI-2 defueling progress, nor should they be construed as a determining

factor for transition to a defueled state as defined in Section 1.3 of the TMI-2 Technical Specifications; transition was based on the TMI-2 Defueling Completion Report which demonstrated the overall adequacy of the defueling progress in the context that inadvertent criticality is no longer possible and defueling has progressed to a point which, based on current technology and ALARA considerations, is defined as the logical endpoint for current defueling activities.

Each PDSR sub-report was submitted to the Nuclear Regulatory Commission (NRC) as measurements, analysis and the internal review and approval process were completed.

#### **PDSR FORMAT**

A standard format was utilized for all PDSR sub-reports. Certain adaptations were required based on the type of measurement techniques applied and the varying types of data generated.

In general, each PDSR sub-report contains the following sections:

- Summary
- Introduction
- Background
- Methods
- Analysis
- Conclusion
- References
- Tables (as necessary)
- Figures and other attachments  
(as necessary)



The Summary section of each sub-report precedes the Introduction section and consists of a brief statement including the estimate of record of the amount of fuel remaining, the physical forms of the fuel, the locations of the fuel, a brief statement of the measurement methods used to measure the fuel, the analytical techniques utilized to determine the estimate of record, and a statement of any pertinent conclusions and recommendations for confirmatory measurements, if appropriate.

The Introduction section of each sub-report explains that the individual sub-report is one in a series of reports generated to fulfill the requirements of the SNM Accountability Program. This section also describes the content of the sections to follow.

The Background section of each sub-report describes the relevant area, system, or component by its physical dimensions and location, the boundaries of the sub-report, the intended function of the area, system, or component, its operations and relationship to the accident and/or subsequent cleanup activities, and how the fuel may have relocated to that area.

The Methods section of each sub-report briefly explains the method(s) utilized to measure fuel in each area, system, or component. Due to the complex structure of the components surveyed, the residual SNM quantity may be determined utilizing a combination of direct measurements, sample analyses, volumetric measurements, and engineering analyses. The sub-reports are supported by detail engineering calculations, vendor reports, and research data.

The Analysis section of each PDSR explains how the "estimate of record" of the amount of remaining fuel was calculated. This section describes the techniques utilized to analyze the available data and discusses supporting data, assumptions, calculations, key observations, various models, and the adequacy of the data. This section also addresses the transport analysis, when applicable, for non-isolated areas, systems, and components.

The Conclusion section of each sub-report states the estimate of record of the amount of fuel remaining in that area, system or component; assigns an appropriate uncertainty; discusses the adequacy and confidence level of the conclusion; and makes a statement defining why the estimate of record is reasonable based upon the available data.

## **METHODS USED TO ASSAY FUEL**

The complex configuration of the TMI-2 plant and restricted access in certain areas increased the difficulty of accomplishing residual fuel assessment at TMI-2. Several different measurement methods were required in order to assess the residual fuel for different areas, systems, and components. Gamma detection, neutron detection, video inspections, radiochemical analysis of samples, and alpha detection were among the methods used to perform this assessment. Brief summaries of these methods are provided below.

Gamma Detection - Gamma detection uses either sodium iodide (NaI) scintillation or high purity germanium (HPGe) detector methods. NaI scintillation is used to detect the Ce-144 2.185

MeV gamma ray. The HPGe detector detects both Ce-144 and Eu-154 gamma rays. These isotopes are suitable tracers for fuel debris at TMI-2 and the quantity of fuel is based on either the Ce-144 to fuel ratio or the Eu-154 to fuel ratio (Reference 5). These ratios change by the radioactive decay of Ce-144 (284.3 days) and Eu-154 (8.8 years). The gamma detection method was generally used wherever the level of background radiation was moderate to low (50 mR hr<sup>-1</sup>).

Neutron Detection - There are two kinds of neutron detection methods: passive counting and active neutron interrogation. The passive counting method detects neutrons directly, employing He-4 detectors for fast neutrons, and He-3 and BF<sub>3</sub> detectors for thermalized neutrons. The active neutron interrogation method detects induced fission neutrons. Sb-Be photoneutrons induce fission reactions. The induced fission neutrons are detected by an He-4 neutron counter. This photoneutron system is calibrated with a known amount of fuel placed at various locations in a mock-up. The neutron detection method is generally used wherever background radiation levels are high in order to confirm measurements made by other methods.

Visual Inspection and Radiochemical Analysis of Samples - If residual deposits are found, visual inspections and debris sampling and analyses may be performed. The quantity of fuel is assessed by debris volume and ratio of fuel to debris. Debris volume is estimated by visual inspections and configuration of components. Ratio of fuel to debris is estimated by debris sampling and analyses. These analyses include volume estimates, weight measurements, and sample analysis for fuel. This method is rarely used alone, but is used in conjunction with other

counting methods in order to help interpret those results.

Alpha Detection - A proportional alpha counter is a gas-filled chamber with two electrodes. This counter detects current between the electrodes in an amount proportional to the amount of ionization. This detector readout assembly is able to select alpha particles from other ionizing radiation by pulse height discrimination. The fuel ratio of alpha activity is 151  $\mu\text{Ci/g}$  of fuel, and has a half-life long enough to require only a small decay correction. The alpha detection method is used only for thin films, due to the short range of an alpha particle.

## FUEL MEASUREMENT UNCERTAINTIES

Where possible, SNM accountability measurements were conducted in areas that had been physically isolated to prevent transport of SNM into or out of the area, system, or component. However, in some cases, it was not possible or feasible to conduct an SNM measurement of an area, system, or component in an isolated configuration. In these cases, the PDSR included an analysis which demonstrated that there could be no significant transport of SNM into or out of the area, system, or component after the completion of the SNM measurement.<sup>1</sup>

Some of the estimate of record quantities reported for residual SNM quantities are referred to -

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<sup>1</sup> An exception to this was the RV draindown to the fuel transfer canal (FTC). No special precautions were taken to prevent fuel transport to the FTC. However, little fuel was in fact transferred.

as the minimum detectable level (MDL). This means that the measurement technique did not detect a statistically significant number of events (counts) related to SNM. Therefore, the true quantity of SNM believed to be in the applicable area, system, or component is equal to or less than the reported MDL. For example, if the residual SNM quantity is reported to be an MDL of 3 kg, the true quantity of residual SNM could be any value from 0 to 3 kg.

Physical measurements of fuel quantities are subject to imprecisions. The accuracy of post-defueling survey SNM measurements was significantly impacted by the inaccessibility of SNM locations, high background gamma radiation dose rates, unknown SNM distribution characteristics and low neutron emission rates for TMI-2 fuel. The precision of the measurements was negatively impacted by the short half life of the preferred high energy fuel tracer isotope Cerium-144 (2.1 MeV gamma emitter) as compared to the lower energy of Europium-154 (1.1 MeV gamma emitter), which is a fuel tracer isotope with a longer half life (i.e., 8.8 years). Uncertainties were also due in part to the combined effects of counting statistics, representativeness of samples to the whole, high radiation interference backgrounds which elevate MDLs, complex and undefined fuel distribution geometries, lack of personnel access requiring use of remote measurement techniques, and varying signal absorption rates due to the presence of structural members and blanketing layers. The uncertainty of the total "estimate of record" tends to be dominated by the largest single measurement aggregate error or largest MDL. Laborious measurements at precision values much smaller than the largest aggregate error would have consumed measurement resources unnecessarily. The principles of ALARA suggest that measurement effort be proportioned according to the absolute uncertainty

of fuel in any measurement location. The analysis of the Reactor Vessel residual fuel quantity yielded an uncertainty of  $\pm 370$  kg. Therefore, state-of-the-art measurements of other reactor coolant and auxiliary system components of much smaller fuel content (e.g., 3 kg) would have included errors that were diminishingly small compared to the overall uncertainty set by the Reactor Vessel. Since the component bias estimate errors were independent, they were combined in quadrature (i.e., the square root of the sum of the squares of the individual biases) to obtain their contribution to the overall error.

Because of the inability to control some of the variables described above, post-defueling survey measurements at TMI-2 were attended by significant uncertainties. These uncertainties were minimized to the extent possible by the judicious selection of measurement techniques and a graded application of resources. Areas, systems, and components known or believed to contain SNM quantities greater than 1 kg were subjected to significantly greater measurement resources than locations where less than 1 kg of SNM was indicated. Nonetheless, the variables which negatively impact precision and accuracy of post-defueling survey SNM measurements resulted in some relatively large uncertainties. The PDSR sub-reports include the estimate of record of residual SNM and associated uncertainty. Therefore, measurement reports contain a calculated uncertainty and/or qualitative discussion of the adequacy of the measurement and identify potential source(s) of measurement error. The use of the term "estimate of record" in this context conveys that the quantity reported is the most realistic value for the quantity of SNM in the area, system, or component based upon the available measurement data.

## REFERENCES

1. NRC letter, "Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d) and 70.53," dated October 17, 1985.
2. GPU Nuclear Administrative Procedure 4000-ADM-4420.02, "TMI-2 Core and Special Nuclear Material Accountability Program."
3. TPO/TMI-189, "Reactor Vessel Post-Defueling Special Nuclear Material Survey," Revision 0, January 1988.
4. TPO/TMI-187, "Instrument Selection for Residual Fuel Measurements," Revision 0, January 1987.
5. GPU Nuclear Technical Bulletin 86-41, "Ce-144, Eu-154, and Eu-155 as Tracers for Fuel Debris," Revision 2, February 1988.
6. GPU Nuclear Policy/Plan Procedure 4000-PLN-4420.02, "SNM Accountability Plan."

## ATTACHMENT 1

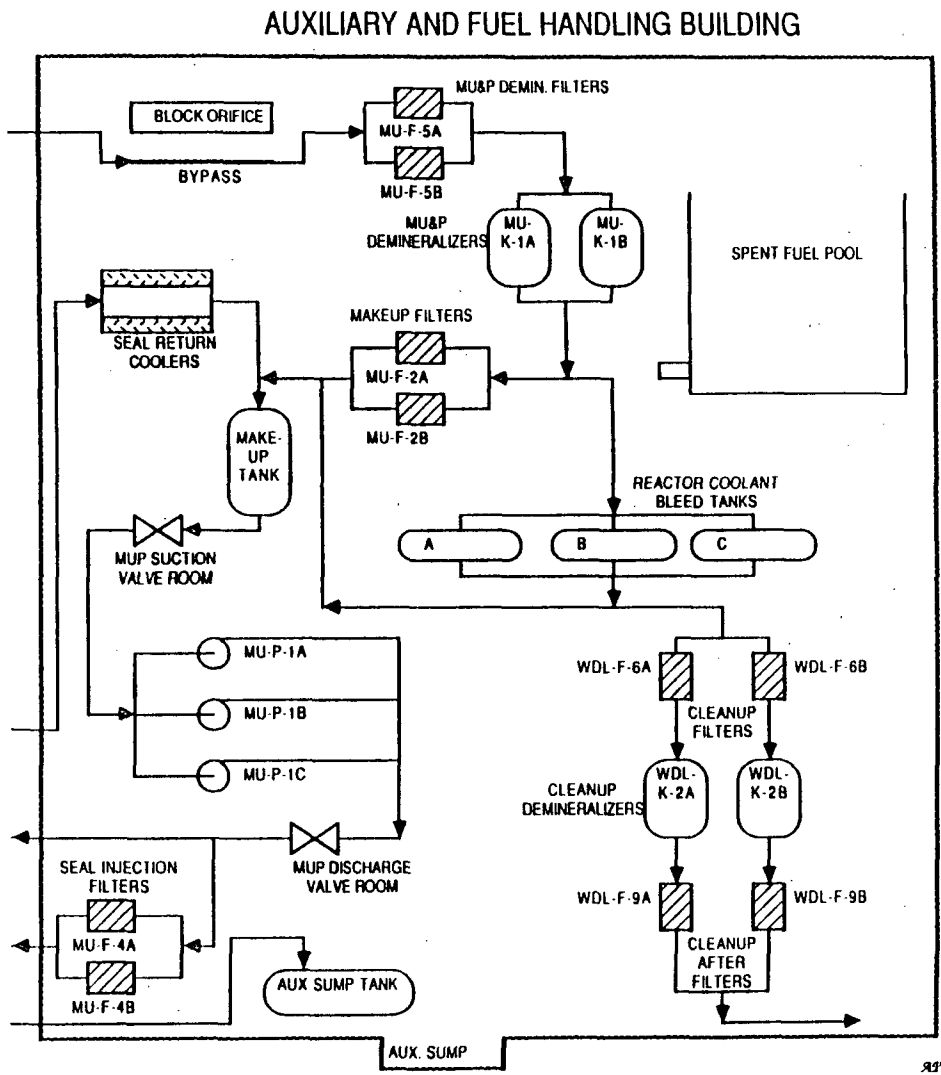
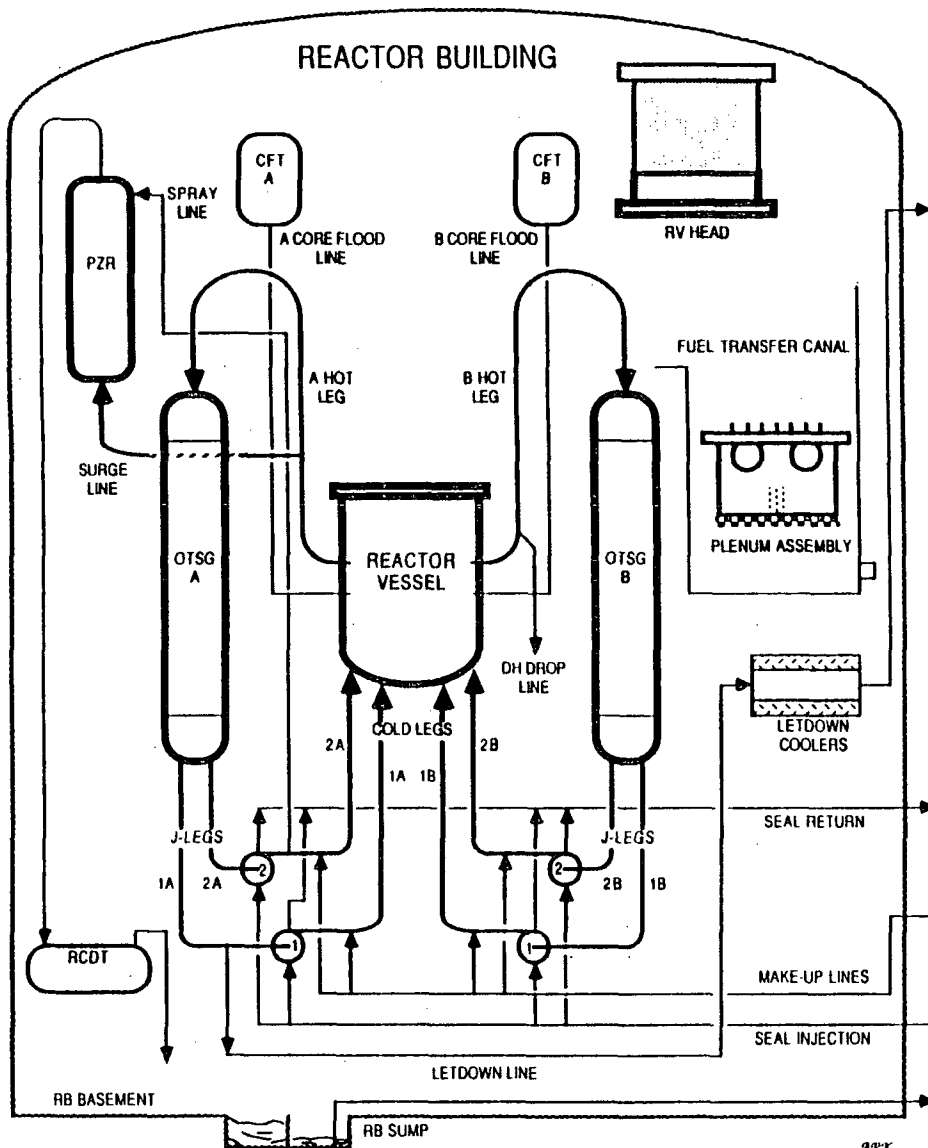
### LIST OF SECTIONS

SECTION TITLE	GPUN LETTER NUMBER	DATE OF SUBMITTAL
Plenum	4410-88-L-0162	09-30-88
Letdown Coolers	4410-89-L-0097	09-22-89
Pressurizer	4410-89-L-0097	09-22-89
Reactor Vessel Head	4410-90-L-0019	03-14-90
Reactor Building Basement	4410-89-L-0097	09-22-89
'A' and 'B' OTSG	C312-92-2064	08-20-91
Auxiliary and Fuel Handling Buildings	C312-91-2045	06-07-91
Reactor Building Miscellaneous Components*	C312-91-2052	06-18-91
Reactor Coolant System**	C312-92-2055	07-03-91
Reactor Vessel	C312-93-2004	02-01-93

\* Includes Incore Guide Tubes in "A" D-ring, Fuel Transfer Canal/Transfer Tubes, Upper Endfittings, Core Flood System, Tool Decon Facility and the Drain Line from the Tool Decon Facility, Temporary RV Filter System, Defueling Water Cleanup System, Reactor Coolant Drain Tank, and Defueling Tools.

\*\* Includes Cold Legs, Hot Legs, Reactor Coolant Pumps, Decay Heat Line, Core Flood Lines, and Pressurizer Lines.





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Buckley, John | Friday, February 10, 2012

Document:  **Rev 1 to "TMI-2 Post-Defueling Rept for A & B Once-Through Steam Generators."** (Version 1.0, Released)

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GPU Nuclear Corporation  
Post Office Box 480  
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717 944-7621  
TELEX 84-2386  
Writer's Direct Dial Number:  
(717) 948-8400

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Three Mile Island Nuclear Station, Unit 2 (TMI-2)  
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SNM Accountability

Dear Sir:

The Post-Defueling Survey Report (PDSR) for the 'A' and 'B' Once-Through Steam Generators (OTSGs) was submitted by GPU Nuclear letter 4410-90-L-0019, dated March 14, 1990. The attached revision to that PDSR was necessitated by the results of analysis of "fluff" samples acquired from the OTSG J-legs and lower heads subsequent to the initial submittal of the PDSR. Included are a revision summary and the revised OTSG PDSR with change bars denoting the revisions.

Sincerely,

R. L. Long  
Director, Corporate Services/TMI-2

EDS/dlb

cc: T. T. Martin - Regional Administrator, Region I  
M. T. Masnik - Project Manager, PDNP Directorate  
L. H. Thonus - Project Manager, TMI  
F. I. Young - Senior Resident Inspector, TMI

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TMI-2 POST-DEFUELING SURVEY REPORT  
FOR  
THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS  
(REVISION 1)

APPROVED: *J. S. [Signature]* 7/24/91  
Manager, TMI-2 Licensing

APPROVED: *J. B. [Signature]* 7/29/91  
Manager, TMI-2 Engineering

TMI-2 INTERNAL REVIEWS

Author:

C. H. Distenfeld 7/15/91  
C. H. Distenfeld

Responsible Technical Reviewer:

R. H. Freeman 7/15/91  
R. H. Freeman

Manager, Fuel Measurements & Analysis:

C. H. Distenfeld 7/15/91  
C. H. Distenfeld

PDSR Committee Chairman:

R. E. Rogan 7/17/91  
R. E. Rogan

**TMI-2 POST-DEFUELING SURVEY REPORT**  
**FOR**  
**THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS**

**(REVISION 1)**

TMI-2 POST-DEFUELING SURVEY REPORT  
FOR  
THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS  
(REVISION 1)

REVISION SUMMARY

The TMI-2 Post-Defueling Survey Report (PDSR) for the 'A' and 'B' Once-Through Steam Generators (OTSGs) has been revised due to the latest fluff sampling acquisition from the OTSG J-Legs and lower heads. The details for this revision are contained in Engineering Calculation 4249-3224-91-007.

The estimate of record of the amount of uranium dioxide ( $UO_2$ ) reactor fuel remaining in the 'A' and 'B' OTSGs has been increased from 4.1 kg to 7.1 kg  $\pm 33\%$  in the 'A' OTSG and from 51.4 kg to 55.2 kg  $\pm 17\%$  in the 'B' OTSG for the following reasons:

1. New samples were taken from the OTSG lower heads and J-Legs which indicate that the fuel debris remaining in the OTSGs is similar to the fuel debris previously taken from the lower head of the reactor vessel.
2. Results from fuel debris samples taken from the lower head of the reactor vessel show more Cs-137 activity per gram of debris than did the fuel debris sample data used in the original calculation.
3. A different fuel debris density and weight percent fuel fraction was also incorporated. The revised values were based on the latest fluff samples taken from the lower head and J-Legs (Reference Engineering Calculation 4249-4400-91-004).



### Revision of the Isotopic Cs-137 Activity

The original calculation used Cs-137 activity data determined for fuel debris samples taken from the pressurizer. The latest fuel samples taken from the OTSGs show a higher activity per gram which results in a greater calculated exposure rate for each Microshield computer run. The original calculation used 3.665 E+2 microcurie per gram for Cs-137 while the new sample data results for the lower head debris indicate 1.1290 E+3 microcuries per gram.

### Revision of the Fuel Debris Density Values

The fuel sample data used in the original calculation were based on a combination of both pressurizer and reactor vessel material densities of 2.0 and 2.5 grams per  $\text{cm}^3$ . The latest sample data taken from the lower head and J-Legs of the OTSGs indicates that the debris material density is closer to 3.2  $\text{g}/\text{cm}^3$ . Therefore, the revised calculation utilized the new material density of 3.2  $\text{g}/\text{cm}^3$  for all locations in the OTSGs.

### Revision of the $\text{UO}_2$ to Fuel Debris Ratio

The latest fluff sample data for fuel debris taken from the J-Legs and lower heads of the OTSGs indicate that the  $\text{UO}_2$  to fuel debris ratio was 0.575 or 57.5% by weight. This new fuel fraction was the average value determined from twenty-eight (28) samples obtained from the OTSGs (Reference GPUNC Calculation 4249-4400-91-004).

### Conclusion

The above changes result in a small increase in the estimate of record for the OTSGs from 55.5 kg to 62.3 kg  $\text{UO}_2$  with an uncertainty of approximately  $\pm 16\%$  (one sigma).

TMI-2 POST-DEFUELING SURVEY REPORT  
FOR  
THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS

## SUMMARY

The estimates of record of the amount of uranium oxide ( $UO_2$ ) remaining in the Three Mile Island Unit 2 (TMI-2) 'A' and 'B' once-through steam generators (OTSGs) are  $7.1 \text{ kg} \pm 33\%$  and  $55.2 \text{ kg} \pm 17\%$ , respectively. All statistical uncertainties are expressed as  $\pm$  one sigma limits for combined increments calculated or taken to be one sigma (defined as one standard deviation). The  $UO_2$  is distributed as follows:

	<u>'A' OTSG</u>	<u>'B' OTSG</u>
Upper Tube Sheet	$1.4 \text{ kg} \pm 21\%$	$36.0 \text{ kg} \pm 18\%$
Tube Bundle	$1.7 \text{ kg} \pm 48\%$	$9.1 \text{ kg} \pm 48\%$
Lower Head	$1.4 \text{ kg}$	$2.2 \text{ kg}$
RCP-1 J-Leg	$1.5 \text{ kg}$	$1.9 \text{ kg}$
RCP-2 J-Leg	$1.1 \text{ kg}$	$6.0 \text{ kg}$
TOTAL	$7.1 \text{ KG} \pm 33\%$	$55.2 \text{ kg} \pm 17\%$

The 'B' upper tube sheet was characterized by neutron activation measurements. Fuel estimates of the 'A' upper tube sheet, the 'A' and 'B' OTSG tube bundles, the lower heads, and the J-legs were projected using gamma radiation measurement data. Earlier fuel estimates based on independent transuranic analysis of surface scrapings (Reference 1) and on direct alpha particle measurements of a limited portion of the 'A' OTSG tube bundle (Reference 2) indicate that the present values are conservatively high.

The quantities of  $UO_2$  remaining in the 'A' and 'B' OTSGs are  $<0.5\%$  and  $<5.5\%$ , respectively, of the anticipated residual  $UO_2$  inventory\* for the entire TMI-2 facility in Mode 2.

Independent foil activation measurements were performed by a Battelle Northwest Laboratory group on both OTSG lower head and J-leg locations. These measurements confirmed the relatively low fuel deposition in these locations (Reference 4).

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\* The anticipated residual  $UO_2$  inventory is as defined in the PDMS Safety Analysis Report and the PDMS Programmatic Environmental Impact Statement and is based on the assumption that the defueling program goal to remove more than 99% of the original core inventory of  $UO_2$  is achieved.

TMI-2 POST-DEFUELING SURVEY REPORT  
FOR  
THE 'A' AND 'B' ONCE-THROUGH STEAM GENERATORS

1.0 INTRODUCTION

This report presents the analysis of the residual inventory of uranium dioxide ( $UO_2$ ) in the Three Mile Island Unit 2 (TMI-2) 'A' and 'B' once-through steam generators. It is one in a series of reports generated to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 5). All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation).

Section 2.0, "Background", describes the physical attributes of the 'A' and 'B' OTSGs and their relationship to the accident and subsequent cleanup activities. The boundaries for this report are also discussed.

Section 3.0, "Methods", describes how fuel measurements and sample data were used to produce the estimates of record. Copper activation foils were primarily used to determine the estimate of record of the amount of fuel on the 'B' OTSG upper tube sheet. Separate copper foil activation calibrations were performed in the 'A' and 'B' OTSG upper heads with a known neutron source (Am-Be). The calibration was used to compensate for uncertain scattering and neutron absorption (Reference 6). Fuel measurement data for the 'A' upper tube sheet, the OTSG tube bundles, the lower heads, and the J-Legs consisted primarily of gamma radiation measurements, which were correlated to actual reactor coolant system (RCS) sample analysis results to produce the estimate of record. The results of using gross gamma measurements to characterize fuel present as films and as tube blockages was compared to earlier fuel analysis by direct alpha measurements and by scrapings taken from the 'A' and 'B' OTSG upper head manway and inspection hole cover plate inserts (Reference 1). Direct alpha film measurements of the top 20 feet of nine 'A' OTSG tubes also were used for comparison (Reference 2).

Section 4.0, "Analysis", explains how the estimates of record of fuel in the 'A' and 'B' OTSGs were calculated based on the fuel measurement and sample analysis data and compared to calculated exposure rates for known activity concentrations modeled with the Microshield computer program (Reference 7).

Section 5.0, "Conclusion", presents the estimates of record for the amount of  $UO_2$  remaining in the 'A' and 'B' OTSGs, and states supporting rationale leading to the conclusion that the estimates are reasonable based upon the available sample analysis data and the fuel measurement techniques employed.

## 2.0 BACKGROUND

The OTSGs are vertical, straight tube and shell boilers in which the reactor coolant (the heat source) is on the tube side and the secondary coolant is on the shell side (Figure 1). They are made of high strength carbon steel and all surfaces that contact the primary coolant are stainless steel or inconel. The TMI-2 'A' and 'B' OTSGs were used for transferring heat from the reactor coolant system to the secondary system.

As a result of the TMI-2 accident, fuel was transported through the hot legs into the OTSGs by a variety of pathway mechanisms which include drain and refill of the steam generator and inlet pipings, forced circulation from the coolant pumps, natural circulation, and fluid movement resulting from energy releases into the reactor vessel (Reference 8).

Fuel fines and sediment material were transported to the OTSGs by reactor coolant through the hot legs to the steam generator upper head and tube sheet. Much of the larger particulate material settled out on the upper tube sheet with the finer material being transported through the tube bundles to the lower head and outlet piping.

This Post-Defueling Survey Report (PDSR) includes characterization of the 'A' and 'B' OTSG upper tube sheets, tube bundle regions, lower heads, and the associated coolant outlet J-legs.

### Radiological Environment

As a result of the March 1979 accident, fuel debris and fission products were released and deposited in the 'A' and 'B' OTSGs. Most of the larger debris material settled out on the upper tube sheets with smaller material being transported to the outlet piping. During late 1987, the debris material on the upper tube sheets was removed by vacuuming and pick-and-place operations. The total person-hours and person-rem expended for defueling the 'A' upper tube sheet was 108 and 4.3, respectively. The total person-hours and person-rem expended for defueling the 'B' upper tube sheet was 153 and 9.4, respectively. Currently, radiation levels in the 'A' OTSG upper tube sheet range from 1-3 R/hr. In the 'B' OTSG, levels range from 30-100 R/hr. The general area exposure rate in the 'A' and 'B' D-rings ranges from 50-800 mR/hr. In general, higher exposure rates and significant loose surface contamination exists in the 'B' D-ring area.

The total person-hours and person-rem expended for the OTSG residual fuel measurement program were:

<u>OSTG Lower Head and J-Legs</u>	<u>Person-hours</u>	<u>Person-rem</u>
'A'	164	7.4
'B'	140	8.5

<u>Upper Tube Sheet</u>	<u>Person-hours</u>	<u>Person-rem</u>
'A'	160	7.1
'B'	348	21.9

The methodology covered by this report is considered to be optimized toward keeping radiation exposure as low as reasonably achievable (ALARA), due to the nature of the 'A' and 'B' OTSGs. Due to the radiologically hostile nature of the OTSGs and their environment, further refinements are not considered to be beneficial in light of the cost of exposure and labor.

### 3.0 METHODS

As part of the Special Nuclear Material (SNM) Accountability Program at TMI-2, gamma measurement surveys and neutron activation measurements were performed for fuel characterization of the 'A' and 'B' OTSG upper tube sheets, tube bundles, lower heads, and associated cold legs during September 1988 to mid-January 1989.

SNM accountability of the OTSGs presented a difficult task due to system inaccessibility and high radiation levels, requiring the selection of indirect fuel measurement techniques. Each entire OTSG, including the upper tube sheets, tube bundle region, lower head, and associated cold legs, required fuel characterization. Due to the inaccessibility and presence of primary coolant in the tube bundle region, lower head, and cold legs, an alternative method for estimating fuel was developed. This method consisted of obtaining gross gamma measurements, correlating this data to predicted gamma exposure rates for fission products sequestered in surface films, and for clustered fuel debris blocking tubes with a composition assumed to be similar to samples collected from the OTSG upper tube sheet, and estimating fuel content. Different types of debris in various geometries were modeled using the Microshield computer program to generate gamma exposure rates based on known isotopic quantities. The actual in situ gamma radiation measurements were then compared with the computer-generated values to estimate fuel content.

## OTSG Characterization

The OTSGs were characterized in two phases. The first included measurements of the lower heads and associated outlet J-legs. The second phase included measurements of the upper tube sheets and the tube bundle region.

### Phase One

During phase one of the measurement program, CoPhysics Corporation, under contract to GPU Nuclear, fabricated several fuel measurement strings to obtain gamma measurements in the 'A' and 'B' OTSG lower heads and associated J-legs. The fuel measurement strings consisted of 100-ft long by 0.5-in. diameter polypropylene tubes which contained, at the front ends, two Geiger-Müller (GM) probes spaced at 6-ft intervals. The fuel measurement strings were deployed through a guide tool mounted on the steam generator upper tube sheet and were pushed down through a 56-ft long steam generator tube (0.56-inch inside diameter) to the lower head. They were then pushed an additional 15 ft further into the associated J-legs. Additionally, miniature lights and a Welch Allyn videoprobe were deployed through adjacent tubes to document placement of the strings within the respective RCS component and within any observed debris.

Each fuel measurement string also contained copper foils for independent measurement of fuel-related material by a DOE national laboratory. The foil analysis was performed and reported by the Pacific Northwest Laboratory (Reference 9).

During deployment of the fuel measurement strings, stabilized assay meters (SAM-2s) were connected to the GM probes to obtain dual gamma radiation readings at every 2 ft in the tube bundle region and at 1-ft intervals in the lower heads and J-legs.

The fuel measurement strings were visually inspected to verify the string positioning and contact with debris material within the steam generator lower heads and J-legs. The debris in the lower heads and J-legs was a



low-density material and was easily suspended while moving the videoprobe which resulted in poor visibility. This low-density material was uniformly distributed with a few areas containing granular material with a higher density and greater fission product content that corresponded well to the higher exposure rates.

## Phase Two

Characterization of the OTSG upper tube sheets in phase two of the measurement program involved placement of copper foils inside the 'A' and 'B' OTSG upper tube sheets to estimate the amount of residual fuel remaining by neutron activation of copper foils. Activation is generated by neutrons emitted from residual fuel. Following activation, the foils were removed and counted in a coincidence system. This system determines the quantity of  $\beta^+$  emissions from the activated copper foils. This process was repeated with an Am-Be neutron source to calibrate the foil and detector system for the scattering and neutron loss environment of the OTSG upper heads. Using the foil activation data, the average neutron activation flux was calculated to determine the fuel estimate of record for the upper tube sheets.

Characterization of the tube bundles for fuel debris blockages involved the deployment of a single measurement string into a total of 52 evenly-spaced steam generator tubes out of 15,531 tubes. The probing locations were selected based on the effective radius of sensitivity of a gross gamma detector probe to a 1-inch long debris blockage. An 8-inch effective horizontal radius was calculated by extensive modeling with the Microshield computer program which resulted in 52 deployment locations needed for full coverage characterization of the 9.5-ft diameter tube bundle region. The one inch blockages were assumed to be vertically displaced 5.5 inches from a given measurement point and always at a horizontal radius of 8 inches. This geometry will maximize fuel estimates by this method since the assumed location is roughly midway between measurement points. The Phase 2 fuel measurement string contained six GM probes at the front end, spaced at 1-ft intervals. The string was deployed into a steam generator tube similar to that used in

phase one, and gamma measurements were recorded at 5-ft intervals down through the total 56-ft tube length.

The gamma readings from all GM probes were collected simultaneously at each 5-ft interval in the tube using 2 IBM personal computers, computer-operated Counter/Timer Boards, LabTech Notebook<sup>TM</sup> software (Reference 10), and preamplifier-amplifier-discriminators (PADs) connected to the GM probes. Two computer systems were installed to enable remote data collection from outside of the TMI-2 Reactor Building. The host computer system was installed in the Reactor Building along with a nuclear instrument module regulated power supply (NIM BIN) containing six PADs and a high-voltage power supply. The remote computer system was installed in the TMI-2 Command Center and connected to the host computer system by standard twisted pair cables and two Black Box Short Haul modems (Reference 11). PC-Anywhere<sup>TM</sup> (Reference 12) software was used to provide remote access to the host computer to operate the data collection system.

The data collection system installed in the host computer included LabTech Notebook<sup>TM</sup> DAS Software and four Metrabyte Counter/Timer Boards<sup>TM</sup> (Reference 13). LabTech Notebook<sup>TM</sup> provided easy-to-use menu-driven data acquisition capabilities along with real time analysis, display, and process control. In our application, LabTech Notebook<sup>TM</sup> was used to collect data using two of the four Metrabyte Counter/Timer Boards<sup>TM</sup>. The Metrabyte Counter/Timer Boards<sup>TM</sup> are short slot boards with 5 independent, 16 bit programmable counters which operate up to 7 MHz for event counting, timing, and frequency counting. The input voltage levels used by these boards are standard TTL level 0 to 5 volts. A universal terminal panel was used to interface with the PADs. The Canberra PADs Model 814A (Reference 14) were internally modified to provide an output pulse amplitude of 3 volts. The counter/timer setup combined with the PADs worked very well.

The fuel measurement strings used in phase two of the measurement program were fabricated, assembled, and calibrated in the Rad Instrument Shop at TMI. The GM probes were connected with RG178U coaxial cable and

installed into polypropylene tubing. At the end of each string, the coaxial cables were terminated into a utility box which provided strain relief for the smaller diameter cables, signal output connection, and a common high-voltage input. Each GM probe in a fuel measurement string was calibrated on a range using a 600 uCi Cs-137 source and a calibration curve was generated correlating exposure rate to counts per minute.

#### 4.0 ANALYSIS

The data collected during phase one of the measurement program was reduced and evaluated and fuel quantities in the 'A' and 'B' OTSG lower heads and J-legs were estimated by comparing the actual in situ exposure rates to calculated exposure rates for known activity concentrations modeled with the Microshield computer program. The gamma radiation profiles for both the 'A' and 'B' OTSG lower heads and J-legs are shown in Figures 2 and 3.

In addition to the gamma radiation measurements performed, visual inspection of the lower heads and J-legs was used to confirm string positioning relative to contact with debris material and for estimating debris depth. The debris observed in the 'A' OTSG was a very low-density material and was easily suspended while moving the videoprobe which resulted in poor visibility. This low-density material was basically uniformly distributed with a few areas that were fluff zones containing material with crust-like surfaces. These areas (the 8 foot locations in Figure 2) corresponded well to the higher dose rates (Reference 15).

The debris observed in the 'B' OTSG lower head was also a very low-density material of minimal depth (less than 1/8 inch). The debris material in the J-legs was fairly uniformly distributed. A few areas exhibited variable debris depths in the RCP '2B' J-leg (the 9 foot location in Figure 3) which contained small gravel-like debris and corresponded to the highest exposure rate (Reference 16).

The amounts of fuel in the 'A' and 'B' OTSG lower heads and J-legs were estimated using several Microshield calculations to model the debris material and geometry in these areas. The composition of the debris was based on analysis of samples taken from the J-legs of both steam generators (Reference 17).

Additionally, for calculation of the source volume, it was assumed (based on Phase 1 visual inspections) that the debris material in the OTSG lower heads was a uniformly distributed low-density material of minimal depth within a 6-foot diameter source. In the J-legs, the debris material was taken to be uniformly distributed (based on Phase 1 visual inspections), with a few areas varying in depth throughout a 9-foot segment of piping (Reference 18). Based on these assumptions, the fuel estimates for the 'A' OTSG lower head and J-legs were 1.4 kg and 2.6 kg, respectively, and for the 'B' OTSG lower head and J-legs were 2.2 kg and 8.0 kg, respectively. Tabulated fuel estimates are shown in Table 1.

The foil activation data collected during phase two of the measurement program for characterization of the 'A' and 'B' OTSG upper tube sheets were evaluated. Four copper foils were placed in the 'B' OTSG and two copper foils were placed in the 'A' OTSG. A copper foil was also exposed in the Reactor Building (RB) above the D-ring to determine the amount of background activation due to environmental neutrons.

The difference in activation flux between the 'B' OTSG upper tube sheet and the RB background gives the activation flux due to residual fuel on the 'B' OTSG upper tube sheet. The estimate of record of residual fuel on the 'B' OTSG upper tube sheet was based on two determinations with resultant fuel estimates of  $36.0 \text{ kg} \pm 18\%$  and  $35.3 \text{ kg} \pm 16\%$  (Reference 6). The 36.0 kg value was chosen to be the estimate of record. This compares to previously determined values by visual-inspection of 28.9 kg (Reference 20) and gross gamma of 53.9 kg (Reference 21).

From the 'A' OTSG foil calibration by the Am-Be source, an effective flux was determined. Based on a comparison of exposure rates between the 'A' and 'B' OTSG upper tube sheets, 3 R/hr and 126 R/hr, respectively (Reference 19), and the 36 kg fuel estimate for the 'B' OTSG upper tube sheet, the 'A' OTSG upper tube sheet fuel estimate of record is 1.4 kg  $\pm$  21%. This value is based on correcting the exposure rate measured at 2 ft above the 'A' OTSG tube sheet to the exposure rate predicted at 1 ft for disc source geometry of the 'B' OTSG tube sheet. The calculated value is 5 R/hr. The ratio of the calculated 'A' OTSG value to the measured 'B' OTSG value of 1 ft separation was used to provide the estimate of record (1.4 kg) from the assayed quantity of 36 kg for the 'B' OTSG.

The measurement data collected for characterization of the 'A' and 'B' OTSG tube bundles was reduced and evaluated to determine the possibility of debris blocking the OTSG tubes. Fuel quantities were estimated by comparing the actual in situ exposure rates to calculated exposure rates based on a one-inch long debris blockage and known activity concentration modeled with the Microshield computer program. It was assumed that the debris material present in a potential blocked tube was most similar in activity and density to the 'B' OTSG upper tube sheet debris material. As previously mentioned, gamma measurements were obtained in the OTSG tube bundles using fuel measurement strings which consisted of six GM probes installed in polypropylene tubing and connected to associated electronics. Calibration response curves were generated for each GM probe in each measurement string. The response characteristics of each GM probe were very uniform and linear within the 100-5000 mR/hr range. Therefore, the response curves for each GM probe in a single string were averaged and calibration surveys and response equations were generated. The use of the averaged calibration data, along with the Lotus 1-2-3 Computer Program<sup>TM</sup> (Reference 22), facilitated the data reduction of approximately 7500 data points.

Probe number one in all fuel measurement strings was shielded with cadmium to eliminate any response to high energy beta and excess response to low energy gamma. Fuel measurement strings were deployed into the

tube bundles in intervals to overlap measurements with probes 1 and 6. The GM probe count data for probes 1 and 6 were reviewed and correction factors for over-response were calculated. These correction factors were applied to probes 2-6 and incorporated into the Lotus 1-2-3 spreadsheet. The 'A' and 'B' OTSG GM probe measurement locations, mean measured exposure rates, and standard deviations are shown in Figures 4 and 5.

Analysis of the 'A' and 'B' OTSG fuel measurement gamma probing data indicates that there are no significant bright spots (i.e., high exposure rates above background) within the tube bundle region that are attributable to fuel blockages. In comparing the radiation profiles for both steam generators, the exposure rates in the dry region of the 'B' OTSG were approximately a factor of 4 higher than the 'A' OTSG. Similarly, the exposure rates in the wet region of the 'B' OTSG were approximately a factor of 2 higher than the 'A' OTSG. High radiation areas within the upper 6 feet of the 'B' OTSG tube bundle are largely due to the contribution from the upper tube sheet. Additionally, high exposure rates were also associated with the water interface, possibly corresponding to radioactive cation capture in the form of a "bath tub" ring. Exposure rates for other areas were relatively uniform and within the calculated deviation.

Previous measurements of fission products and transuranics were performed (Reference 1) on the inside stainless steel surfaces of one upper head nominal 16 inch manway insert for each OTSG. The results (Reference 23) were used to predict that 0.2 and 2.2 kg of  $UO_2$  could be evenly plated out in films in the 'A' and 'B' OTSGs, respectively. Direct alpha measurements (Reference 2) of the top 20 feet of nine roughly evenly-spaced 'A' OTSG tubes were performed. Based on the alpha measurements, approximately 0.09 kg of  $UO_2$  were predicted to be held on all internal surfaces of the 'A' OTSG tubes. This value is in reasonably good agreement with the earlier projection based on the 'A' OTSG manway insert plate.

The second potential fuel compartment, namely fuel debris in blocked OTSG tubes, was evaluated using gross gamma measurements within approximately

evenly-spaced tubes in each generator. The gamma field within a tube is taken to be the sum of background and of possible fuel debris located close enough to be measured. Background includes cosmic ray, contamination external to the OTSG, contamination on the probe string (Figure 6), Cs-137 activity on the inside surface of the tubes, and the 0.07 uCi/cc Cs-137 activity in the primary water wetting the lower portion of the tubes. The Cs-137 activity per unit stainless steel surface (Reference 1), reduced for the lower corrosiveness of the inconel tube material, was used to predict the general exposure rate within the tube bundles from the tubes (Reference 23). The values are approximately 670 and 375 mR/hr for the 'A' OTSG dry and wet parts and 1185 and 705 mR/hr for the 'B' OTSG dry and wet parts, respectively. Using these background values and the contamination on the probes and in the water, no statistically significant positive values in the 'A' and in the 'B' OTSGs were indicated.

The inferential method adopted for determining fuel is a modification of the conventional calculation for the lower limit of detection (LLD). Normally, the LLD is calculated for 95% confidence. A smaller confidence level of approximately 60% was used to partly compensate for two geometry assumptions that lead to unlikely maximum values. They are:

1. Blocked tubes only occur at the maximum effective radius of 8 inches.
2. The point of blockage is always vertically displaced from a measurement location by 5.5 inches.

From earlier discussions, all or nearly all of the gross gamma fields within the OTSGs were due to Cs-137 activity on the primary side surfaces. A leadscrew taken from the TMI-2 reactor vessel head was characterized for adherent films (Reference 24). Variation of factors of 2-3 are reported within distances of a few feet. Due to this expected variability, the OTSG gamma measurements were averaged for each five-foot increment of insertion. Standard deviations were calculated for the variances between the 52 tube results for given insertions,

$\sigma_{(A-L)}$ . The estimate of record was based on converting the  $\sigma_{(A-L)}$  values to  $UO_2$  weight (Reference 25), multiplying by 52 to account for all space at a given depth of insertion, and summing the results for insertion increments A to L as:

$$UO_2 = 52 \sum_{i=A}^L \sigma_i = \text{estimate of record.}$$

The amount of  $UO_2$  predicted to be present in the tubes as films was not added to the total since it was assumed that the gross gamma method already accounted for this increment.

A large percentage of the variation in the rate could be attributed to variations in corrosion layer thickness largely containing Cs-137. It was assumed that all variances are due to fuel.

Based on the previously stated assumptions, the modeled steam generator fuel debris blockage, and the corresponding dose rates, fuel estimates for the 'A' and 'B' OTSG tube bundles were calculated. The estimates of record of total fuel debris deposited in the 'A' and 'B' OTSG tube bundles are 1.7 kg and 9.1 kg, respectively (Reference 25).

## 5.0 CONCLUSION

The use of the fuel measurement strings for obtaining gamma measurements and the copper foils for neutron activation proved to work very well considering the rough handling and strenuous work conditions encountered



in the Reactor Building. Overall, the fuel measurement strings facilitated the deployment of the GM probes into the difficult-to-access areas of the OTSGs and were less dose-intensive to personnel deploying equipment and performing measurements than alternate methods such as probing statistically significant tube quantities of approximately 7000 tubes per OTSG. The estimates of record of the total amount of  $UO_2$  remaining in the 'A' and 'B' OTSGs are approximately  $7.1 \text{ kg} \pm 33\%$  and  $55.2 \text{ kg} \pm 17\%$ , respectively. The uncertainty associated with these fuel estimates (except for the 'B' OTSG upper tube sheet foil activation fuel estimate of  $36.0 \text{ kg} \pm 18\%$ ) is based on modeling geometry errors, source calibration errors, GM probe response errors, and cesium-fuel ratio errors. The total uncertainty is based on the square root of the sum of the squares of the individual uncertainties associated with each component of the analysis. The measurement uncertainty associated with the above fuel estimates varies from 18 to 55% depending on the component.

This estimate of record is derived from existing measurement and sample analysis data. It is expected to remain static since it is not expected that additional quantities of water from the reactor vessel will be added or cycled through the OTSGs.

PS 14

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TABLE 1  
ONCE-THROUGH STEAM GENERATOR  
FUEL ESTIMATES OF RECORD

	<u>'A' OTSG</u>	<u>'B' OTSG</u>
Upper Tube Sheet	1.4 kg ± 21%	36.0 kg ± 18%
Tube Bundle	1.7 kg ± 48%	9.1 kg ± 48%
Lower Head	1.4 kg )	2.2 kg )
RCP-1 J-Leg	1.5 kg ) ± 55%	1.9 kg ) ± 55%
RCP-2 J-Leg	1.1 kg )	6.0 kg )
TOTAL	7.1 kg ± 33%	55.2 kg ± 17%

FIGURE 1  
TMI-2 ONCE-THROUGH STEAM GENERATOR

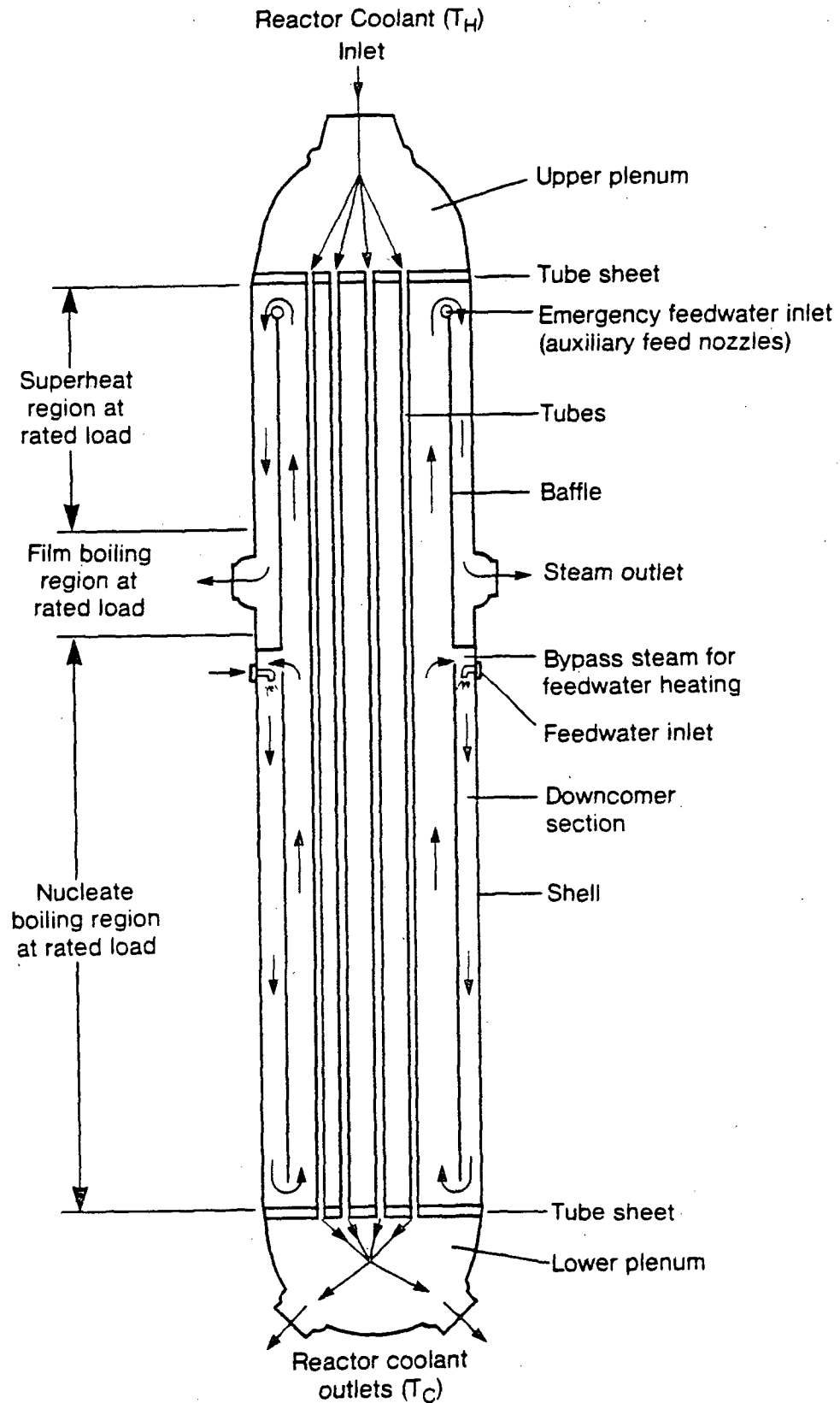
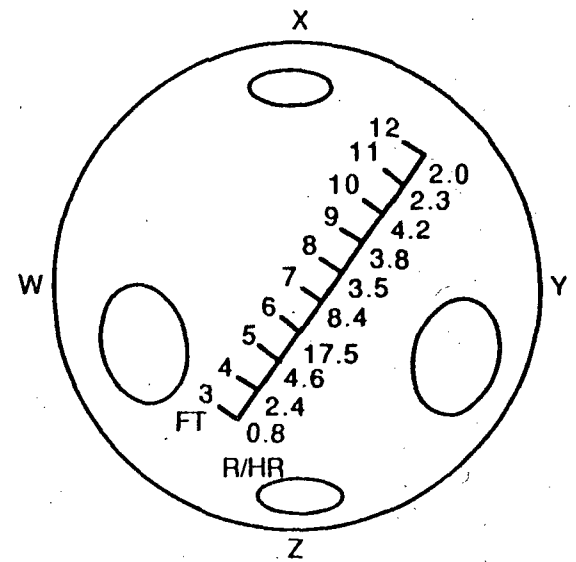
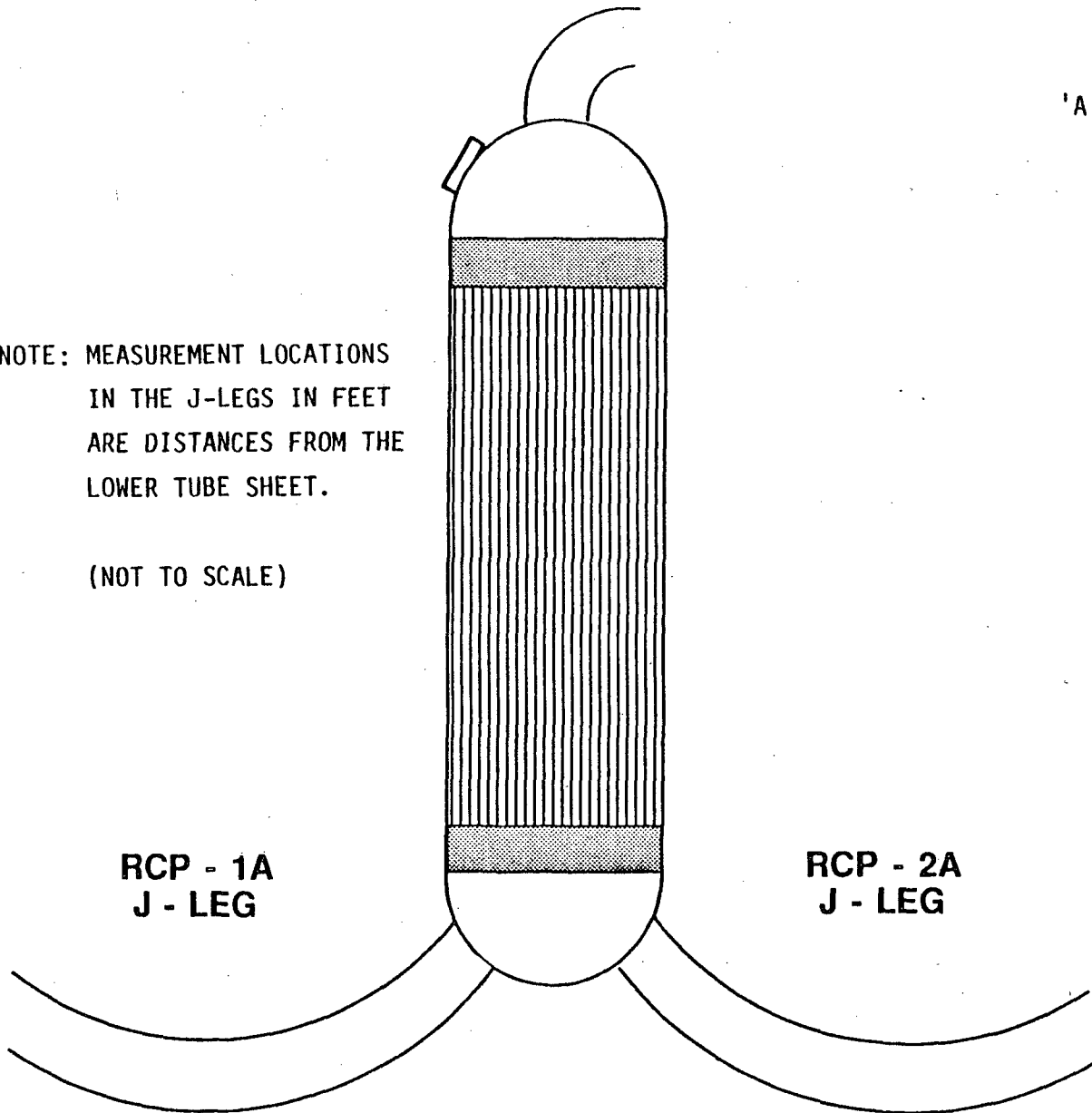


FIGURE 2  
 'A' OTSG LOWER HEAD AND J-LEG  
 EXPOSURE RATE PROFILES

NOTE: MEASUREMENT LOCATIONS  
 IN THE J-LEGS IN FEET  
 ARE DISTANCES FROM THE  
 LOWER TUBE SHEET.

(NOT TO SCALE)



LOWER HEAD - PLAN VIEW

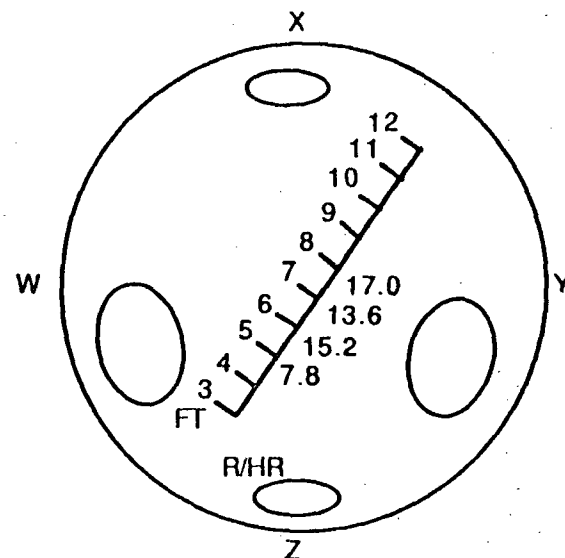
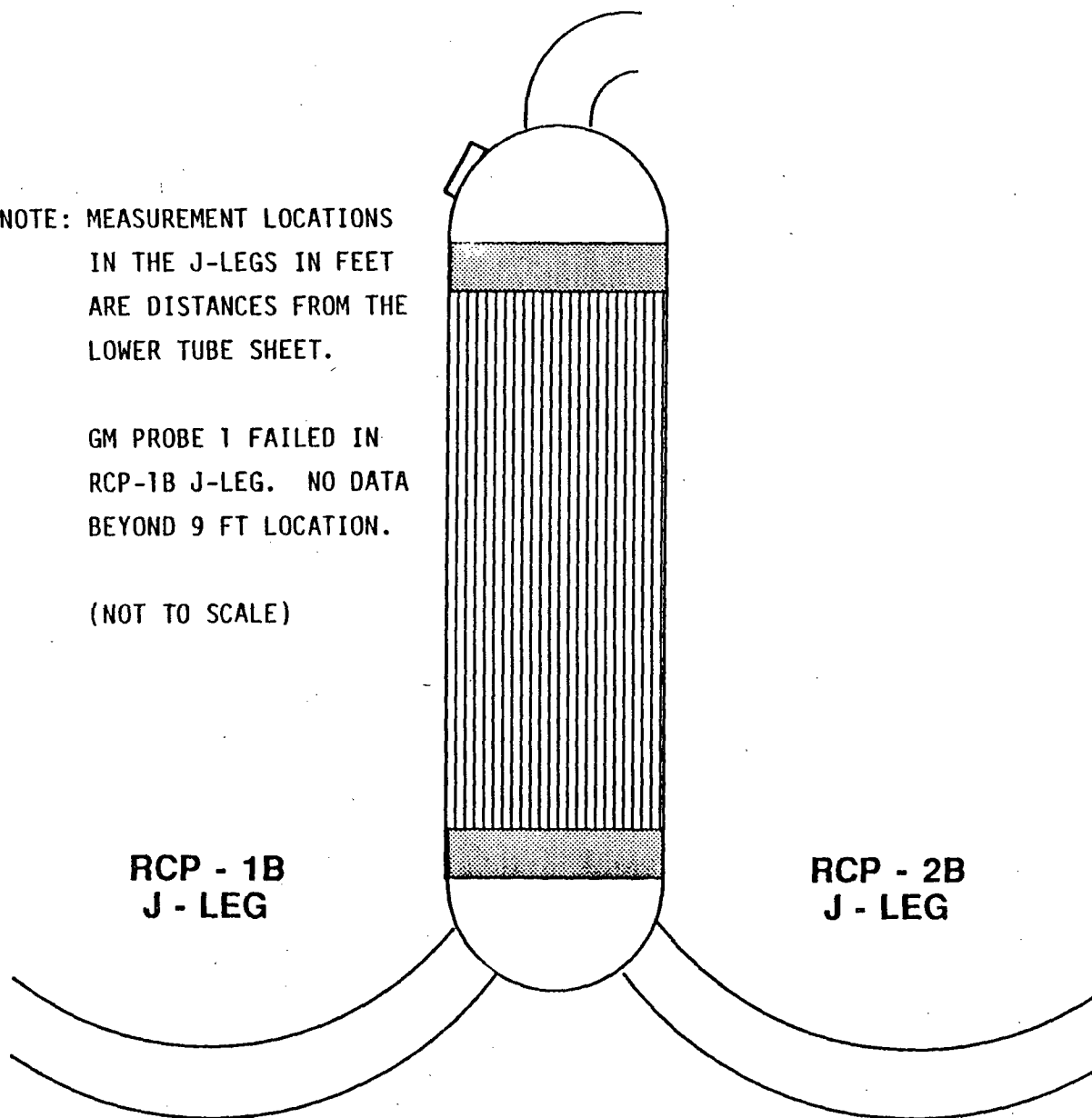
RCP - 1A J - LEG										RCP - 2A J - LEG											
15	14	13	12	11	10	9	8	7	6	5	5	6	7	8	9	10	11	12	13	14	15
FEET																					
DOSE RATE (R/HR)																					
37.5	84	16.3	10.2	18	40	31.5	4.9	1.7	1.5	0.7	0.7	0.8	1.6	3.9	16.5	7.1	5.3	5.8	16.0	76.0	15.0

FIGURE 3  
 'B' OTSG LOWER HEAD AND J-LEG  
 EXPOSURE RATE PROFILES

NOTE: MEASUREMENT LOCATIONS  
 IN THE J-LEGS IN FEET  
 ARE DISTANCES FROM THE  
 LOWER TUBE SHEET.

GM PROBE 1 FAILED IN  
 RCP-1B J-LEG. NO DATA  
 BEYOND 9 FT LOCATION.

(NOT TO SCALE)



LOWER HEAD - PLAN VIEW

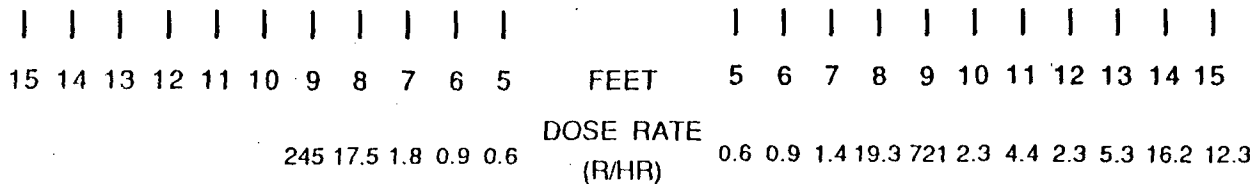
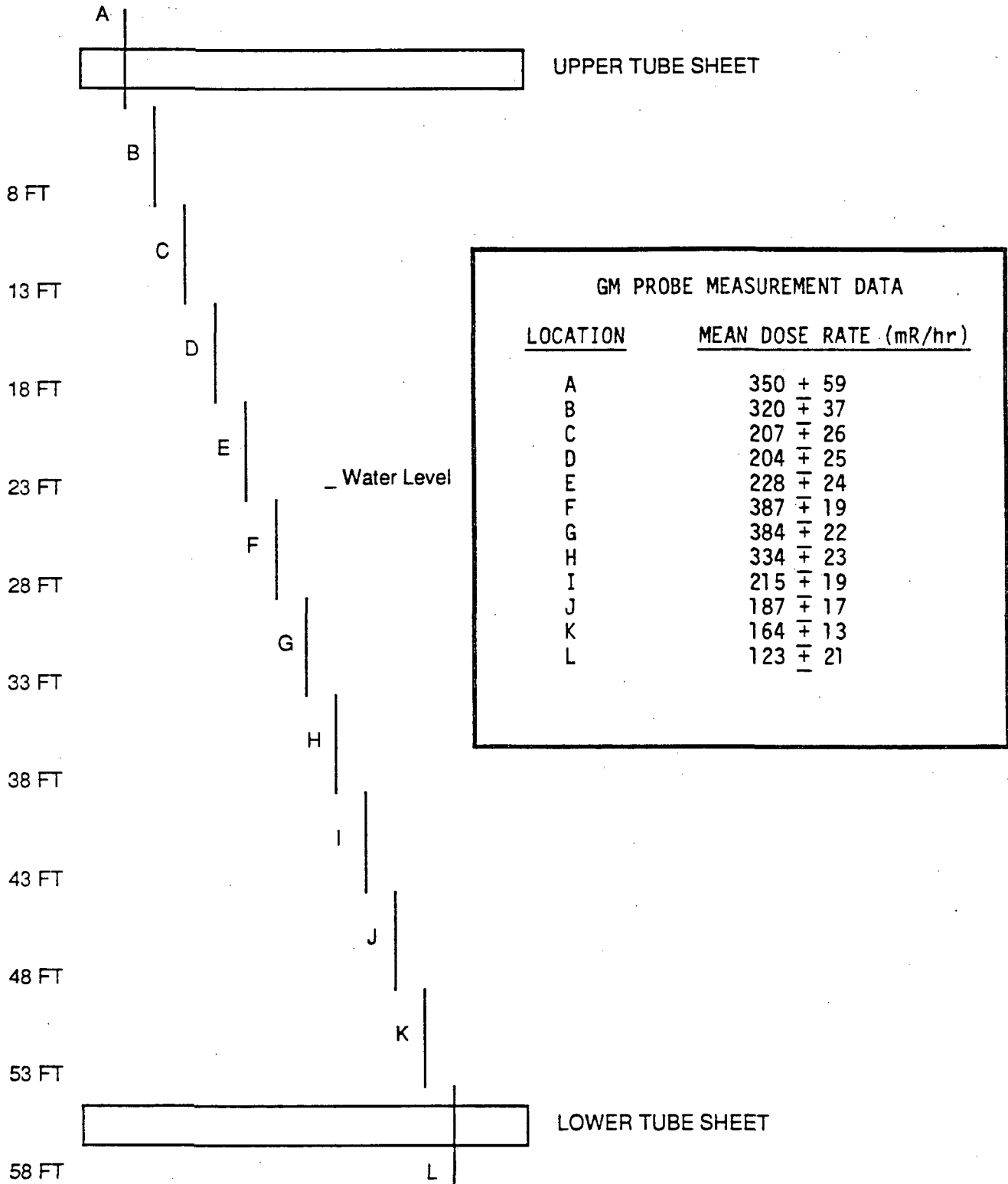


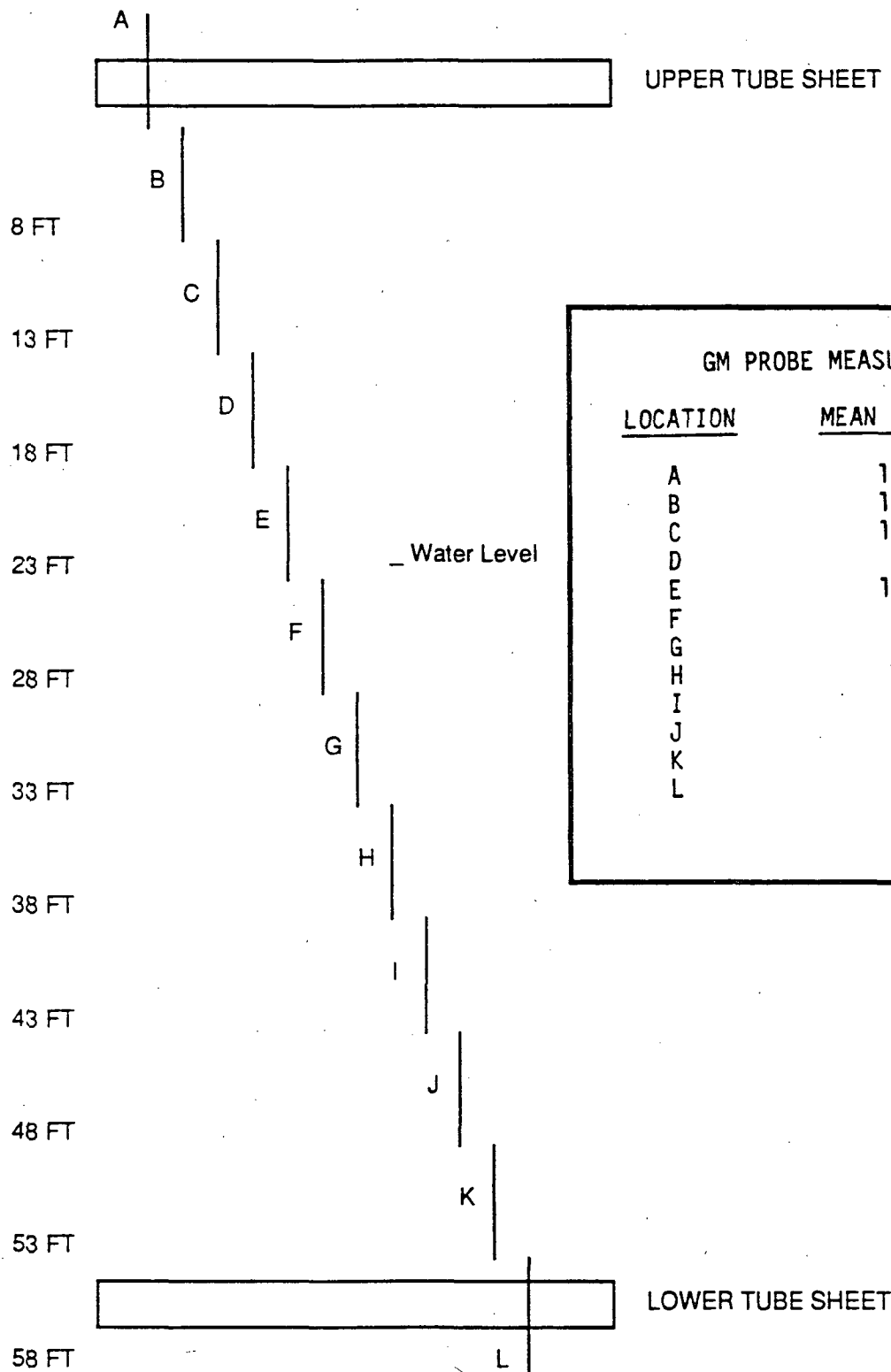


FIGURE 4  
 'A' OTSG TUBE BUNDLE  
 GM PROBE MEASUREMENT LOCATIONS



NOTE: Numbers in feet are the distances below the top surface of the upper tube sheet.

FIGURE 5  
 'B' OTSG TUBE BUNDLE  
 GM PROBE MEASUREMENT LOCATIONS



GM PROBE MEASUREMENT DATA	
<u>LOCATION</u>	<u>MEAN DOSE RATE (mR/hr)</u>
A	1290 + 295
B	1270 + 244
C	1213 + 247
D	679 + 150
E	1311 + 288
F	771 + 124
G	606 + 50
H	540 + 46
I	354 + 27
J	352 + 25
K	359 + 24
L	318 + 28

NOTE: Numbers in feet are the distances below the top surface of the upper tube sheet.

FIGURE 6  
GAMMA PROBE CONTAMINATION

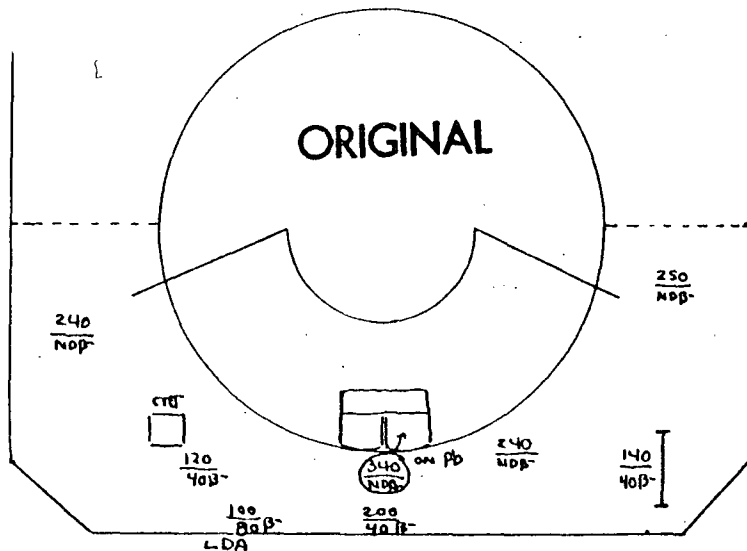
**GPU Nuclear**

RADIOLOGICAL SURVEY

TMI-1  TMI-2  OC

Survey Information	Instrument Data		Air Sample <input checked="" type="checkbox"/>	File Code Number: RB 700 88J 25693
Location 347' B D-RING	Contamination Survey		Date N/T	Smearable Contamination
UNIT 2 R.B.	Inst N/T	Inst N/T	Activity	
Reason	S/N	S/N	Sample #	Location
Job coverage report	Cal. Due	Cal. Due	Note: Radiation dose rates in mr/hr are general area, contamination results are dpm/100cm <sup>2</sup> unless otherwise noted.	βγ
Date 12-21-88 Time 13:25	Eff.	Eff.		α
Tech. G. H. H. G. H. H.	Bkg	Bkg		
Survey Request # N/A	Tech	Tech G. H. H. G. H. H.		
Reviewed by W. Norman	Note: Contact Readings Circled			
		<input type="checkbox"/> Smear Location		


K. Norman DEC 22 1988



- Gamma probe 1<sup>st</sup> TUBE  
100 / 700β
- 2<sup>nd</sup> TUBE  
100 / 700β
- 3<sup>rd</sup> TUBE  
140 / 720β
- 4<sup>th</sup> TUBE  
100 / 220β
- 5<sup>th</sup> TUBE  
80 / 340β
- 6<sup>th</sup> TUBE  
100 / 440β

Remarks:

Buckley, John | Friday, February 10, 2012

Document:  **Exemption from 10CFR30.51,40.61,70.51(d) & 70.53 requirements,per 850418 request.Exemption from 10CFR70.53 shall expire following completion of defueling effort, including assessment of fuel fines & debris within plant.** (Version 1.0, Released)

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Property	Value
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October 17, 1985

Docket No. 50-320

Mr. F. R. Standerfer  
Vice President/Director  
Three Mile Island Unit 2  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, PA 17057

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Docket No. 50-320  
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RCook (Site)  
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LChandler, ELD  
IE (5)  
TBarnhart (4)  
LSchneider

Dear Mr. Standerfer:

Subject: Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d),  
and 70.53

JSaltzman  
ACRS (16)  
Eisenhut/  
Denton

We have reviewed your request, dated April 18, 1985, for exemptions from the requirements of 10 CFR 30.51, 40.61, 70.51(d), 70.53 and 70.54 regarding record keeping, inventorying, and reporting of core special nuclear, source and byproduct materials. As discussed in the attached Exemption, we have determined that you will have sufficient information to comply with the requirements of 10 CFR 70.54 and that an exemption from this regulation is unnecessary. However, we conclude that your request for exemptions from the other regulations are appropriate and acceptable, as stated in the attached Exemption issued by the Director of the Office of Nuclear Reactor Regulation. An environmental assessment of the action considered and a Federal Register notice for this issuance are also enclosed.

Sincerely,

*Original signed by*  
B. J. Snyder

8510230259 851017  
PDR ADOCK 05000320  
P PDR

Bernard J. Snyder, Program Director  
Three Mile Island Program Office  
Office of Nuclear Reactor Regulation

Enclosures:

1. Exemption
2. Environmental Assessment and Notice of Finding of No Significant Environmental Impact
3. Federal Register Notices

cc: T. F. Demmitt  
R. E. Rogan  
S. Levin  
W. H. Linton  
J. J. Byrne  
A. W. Miller

FFSLB:DS:NMSS  
WBBrown\*  
9/30/85

D:DS:NMSS  
RFBurnett\*  
10/3/85

FCMS:NMSS:D  
RECunningham\*  
10/3/85

\*See Previous Concurrence

OFFICE	TMIPO:NRR	Service Distribution List	PD:TMIPO:NRR	ELD	DD:NRR	D:NRR
IRNAME	RWeller;bg*	(see attached)	BJSnyder*	LChandler*	DGEisenhut	HRDenton*
DATE	9/18/85		9/18/85	9/25/85	10/ /85	10/8/85

Docket No. 50-320

Mr. F. R. Standerfer  
Vice President/Director  
Three Mile Island Unit 2  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, PA 17057

Dear Mr. Standerfer:

Subject: Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d),  
and 70.53

We have reviewed your request, dated April 18, 1985, for exemptions from the requirements of 10 CFR 30.51, 40.61, 70.51(d), 70.53 and 70.54 regarding record keeping, inventorying, and reporting of core special nuclear, source and byproduct materials. As discussed in the attached Exemption, we have determined that you will have sufficient information to comply with the requirements of 10 CFR 70.54 and that an exemption from this regulation is unnecessary. However, we conclude that your request for exemptions from the other regulations are appropriate and acceptable, as stated in the attached Exemption issued by the Director of the Office of Nuclear Reactor Regulation, ~~and the Director of the Office of Nuclear Material Safety and Safeguards.~~ An environmental assessment of the action considered and a Federal Register notice for this issuance are also enclosed. *W/B 10/2/85*

Sincerely,

Bernard J. Snyder, Program Director  
Three Mile Island Program Office  
Office of Nuclear Reactor Regulation

Enclosures:

1. Exemption
2. Environmental Assessment and Notice of Finding of No Significant Environmental Impact
3. Federal Register Notices

*pending correction of cover letter as shown above!*

cc: T. F. Demmitt  
R. E. Rogan  
S. Levin  
W. H. Linton  
J. J. Byrne

FFSB:DS:NMSS WBrown 9/30/85	FFSB:DS:NMSS RFBurnett 10/3/85	FCMS:NMSS:D RE Cunningham 10/3/85	D:NMSS JGDavis 9/1/85
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IRNAME ▶	RAWeller;bg	Service Distribution List	B. Snyder	LChandler	DGE:senhut	HR Denton
DATE ▶	9/18/85	(see attached)	9/18/85	9/ /85	9/ /85	9/ /85

## TMI-2 SERVICE LIST

Dr. Thomas Murley  
Regional Administrator, Region 1  
U.S. Nuclear Regulatory Commission  
631 Park Avenue  
King of Prussia, PA 19406

John F. Wolfe, Esq., Chairman,  
Administrative Judge  
3409 Shepherd St.  
Chevy Chase, MD. 20015

Dr. Oscar H. Paris  
Administrative Judge  
Atomic Safety and Licensing  
Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Dr. Frederick H. Shon  
Administrative Judge  
Atomic Safety and Licensing  
Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Karin W. Carter  
Assistant Attorney General  
505 Executive House  
P.O. Box 2357  
Harrisburg, PA 17120

Dr. Judith H. Johnsrud  
Environmental Coalition on  
Nuclear Power  
433 Orlando Ave.  
State College, PA 16801

George F. Trowbridge, Esq.  
Shaw, Pittman, Potts and  
Trowbridge  
1800 M. St., NW.  
Washington, D.C. 20036

Atomic Safety and Licensing Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Secretary  
U.S. Nuclear Regulatory Commission  
ATTN: Chief, Docketing & Service Branch  
Washington, D.C. 20555

Mr. Larry Hochendoner  
Dauphin County Commissioner  
P.O. Box 1295  
Harrisburg, PA 17108-1295

John E. Minnich, Chairperson,  
Dauphin County Board of Commissioners  
Dauphin County Courthouse  
Front and Market Streets  
Harrisburg, PA 17101

Dauphin County Office of Emergency  
Preparedness  
Court House, Room 7  
Front & Market Streets  
Harrisburg, PA 17101

U.S. Environmental Protection Agency  
Region III Office  
ATTN: EIS Coordinator  
Curtis Building (Sixth Floor)  
6th & Walnut Streets  
Philadelphia, PA 19106

Thomas M. Gerusky, Director  
Bureau of Radiation Protection  
Department of Environmental Resources  
P.O. Box 2063  
Harrisburg, PA 17120

Dan Kennedy  
Office of Environmental Planning  
Department of Environmental Resources  
P.O. Box 2063  
Harrisburg, PA 17120

Willis Bixby, Site Manager  
U.S. Department of Energy  
P.O. Box 88  
Middletown, PA 17057-0311

David J. McGoff  
Division of Three Mile Island Programs  
NE-23  
U.S. Department of Energy  
Washington, D.C. 20545

William Lochstet  
104 Davey Laboratory  
Pennsylvania State University  
University Park, PA 16802

Randy Myers, Editorial  
The Patriot  
812 Market St.  
Harrisburg, PA 17105

Robert B. Borsum  
Babcock & Wilcox  
Nuclear Power Generation Division  
Suite 220  
7910 Woodmount Ave.  
Bethesda, MD. 20814

Michael Churchhill, Esq.  
PILCOP  
1315 Walnut St., Suite 1632  
Philadelphia, PA 19107

Linda W. Little  
5000 Hermitage DR.  
Raleigh, NC 27612

Marvin J. Lewis  
6504 Bradford Terrace  
Philadelphia, PA 19149

Jane Lee  
183 Valley Rd.  
Etters, PA 17319

J.B. Liberman, Esquire  
Berlack, Israels, Liberman  
26 Broadway  
New York, NY 10004

Walter W. Cohen, Consumer Advocate  
Department of Justice  
Strawberry Square, 14th Floor  
Harrisburg, PA 17127

Edward D. Swartz  
Board of Supervisors  
Londonderry Township  
RFD #1 Geyers Church Rd.  
Middletown, PA 17057

Robert L. Knupp, Esquire  
Assistant Solicitor  
Knupp and Andrews  
P.O. Box P  
407 N. Front St.  
Harrisburg, PA 17108

John Levin, Esquire  
Pennsylvania Public Utilities Comm.  
P.O. Box 3265  
Harrisburg, PA 17120

Mr. Edwin Kintner  
Executive Vice President  
General Public Utilities Nuclear Corp.  
100 Interpace Parkway  
Parsippany, NJ 07054

Ad Crable  
Lancaster New Era  
8 West King Street  
Lancaster, PA 17602



## UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION	)	Docket No. 50-320
(Three Mile Island Nuclear Station Unit 2)	)	

EXEMPTION

## I.

GPU Nuclear Corporation, Metropolitan Edison Company, Jersey Central Power and Light Company and Pennsylvania Electric Company (collectively, the licensee) are the holders of Facility Operating License No. DPR-73, which has authorized operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2) at power levels up to 2772 megawatts thermal. The facility, which is located in Londonderry Township, Dauphin County, Pennsylvania, is a pressurized water reactor previously used for the commercial generation of electricity.

By Order for Modification of License, dated July 20, 1979, the licensee's authority to operate the facility was suspended and the licensee's authority was limited to maintenance of the facility in the present shut-down cooling mode (44 Fed. Reg. 45271). By further Order of the Director, Office of Nuclear Reactor Regulation, dated February 11, 1980, a new set of license requirements was imposed to reflect the post-accident condition of the facility and to assure the continued maintenance of the current safe, stable, long-term cooling condition of the facility (45 Fed. Reg. 11292). The license provides, among other things, that it is subject to all rules, regulations and Orders of the Commission now or hereafter in effect.

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II.

By letter dated April 18, 1985, the licensee requested exemptions from 10 CFR 30.51, 40.61, 70.51(d), 70.53, and 70.54 regarding the requirements for record keeping, inventorying, and reporting of core special nuclear, source and byproduct materials. Specifically, 10 CFR 30.51 and 40.61 specify the requirements for keeping records which show the receipt, transfer and disposal of source and byproduct material. 10 CFR 70.51(d) specifies the requirements for the periodic conduct of a physical inventory of all special nuclear material in possession. 10 CFR 70.53 specifies the requirements for the periodic submittal of a Material Balance Report and Physical Inventory Listing of special nuclear material possessed by the licensee. 10 CFR 70.54 specifies the requirements for submitting Nuclear Material Transaction Reports for the transfer or receipt of special nuclear material. In meetings with the licensee held subsequent to the April 18, 1985 exemption request, staff representatives of the NRC and Department of Energy (DOE) have determined that the licensee will have sufficient information to comply with the requirements of 10 CFR 70.54 and that an exemption from this regulation is not necessary.

III.

The accident at Three Mile Island Unit 2 severely damaged the reactor core. Video inspections and topography measurements indicate a cavity in the upper core region which represents approximately 26% of the total original core volume. No more than 2 of the original 177 core fuel assemblies

remain intact and only 42 assemblies have any full length fuel rods. The core damage extends radially all the way out to the core former walls. As a result of the accident induced embrittlement of virtually all fuel rods, no fuel assemblies are expected to be withdrawn intact. There is a significant amount of core debris in ex-core region locations (e.g., an estimated 10 to 20 tons in the lower reactor vessel head) and much of the core byproduct material has been released from the fuel. For example, analyses of core debris bed samples indicate that, on the average, only about 13% of the original Cs-137 inventory remains in the fuel although the percentage retained can vary considerably from sample to sample.

During the defueling of the damaged core, the fuel debris will be collected in canisters by vacuuming or "pick and place" techniques. However, as a result of the damaged condition of the core, the licensee will have no means of accurately characterizing (e.g., U-235 enrichment and total uranium content, fission product radionuclide content and distribution, plutonium content) the fuel debris during the defueling sequence. The capability for characterizing the collected fuel debris in each canister would require sophisticated hot cell and laboratory facilities with the means to homogenize, sample, weigh, and analyze the contents of each canister. Such facilities do not exist at Three Mile Island. Given the damaged condition of the core and lack of sophisticated hot cell and laboratory facilities, there is no practical means for the licensee to perform the measurements or precise calculations necessary to comply with the Commission's regulations related to accountability of special nuclear,

source and byproduct materials. The staff therefore concludes that exemptions from the requirements of 10 CFR 30.51, 40.61, 70.51(d), and 70.53 are appropriate. As previously stated in Section II of this evaluation, staff representatives of the NRC and DOE have determined that the licensee will have sufficient information to comply with the transfer requirements of 10 CFR 70.54 and that exemption from this regulation is not necessary.

The granting of these exemptions does not mean that the licensee will not provide any record keeping or reporting of the canister core debris which is intended to be transferred to the custody of the DOE for research and/or storage at DOE facilities in Idaho. In lieu of the reporting requirements of 10 CFR 70.53, the licensee will provide to the DOE all available information describing the physical contents of each canister including: the canister identification number, canister type (i.e., knockout, fuel, or filter), date of shipment, the shipment number, the empty weight of the canister, the loaded weight of the canister, the dewatered weight of the canister, maximum total curies, the canister pressure, general physical description of the canister contents including videotape data (if available), and any additional information based on mutual agreement between the licensee and the DOE. Further, following the completion of defueling and the offsite shipment of the packaged fuel debris, the licensee will be in a position to comply with the requirements of 10 CFR 70.53 and the licensee will be required to submit a Material Balance Report and Physical Inventory Listing at that time.

In lieu of the requirement in 10 CFR 70.51(d) for the periodic conduct of a physical inventory of all special nuclear material, the licensee will conduct such an inventory upon the completion and analysis of a post-defueling survey.

In lieu of the record keeping requirements of 10 CFR 30.51 and 40.61, the licensee will maintain records of each fuel shipment in accordance with the requirements of 10 CFR 71.91. Such records will include an identification of the shipment packaging, the maximum total curies, the total quantity of each shipment, and the date of shipment.

#### IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 30.11, 40.14, and 70.14, these exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The Commission hereby grants exemptions from the requirements of 10 CFR 30.51, 40.61, 70.51(d), and 70.53. The exemption from 10 CFR 70.53 shall expire following the completion of the defueling effort, including an assessment of any fuel fines and debris which remain within the plant, and the subsequent offsite shipment of all packaged fuel debris.

It is further determined that the exemptions do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. In light of this

determination and as reflected in the Environmental Assessment and Notice of Finding of No Significant Environmental Impact prepared pursuant to 10 CFR 51.21 and 51.30 through 51.32, issued September 20, 1985, it was concluded that the instant action is insignificant from the standpoint of environmental impact and an environmental impact statement need not be prepared.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "HR Denton". The signature is written in a cursive, somewhat stylized font.

Harold R. Denton, Director  
Office of Nuclear Reactor Regulation

Effective Date: October 17, 1985  
Dated at Bethesda, Maryland  
Issuance Date: October 17, 1985

UNITED STATES NUCLEAR REGULATORY COMMISSION  
GENERAL PUBLIC UTILITIES NUCLEAR CORPORATION  
DOCKET NO. 50-320  
ENVIRONMENTAL ASSESSMENT AND NOTICE OF FINDING  
OF NO SIGNIFICANT ENVIRONMENTAL IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is planning to issue an Exemption from certain regulations relative to the Facility Operating License No. DPR-73, issued to General Public Utilities Nuclear Corporation (the licensee), for operation of the Three Mile Island Nuclear Station, Unit 2 (TMI-2), located in Londonderry Township, Dauphin County, Pennsylvania.

ENVIRONMENTAL ASSESSMENT

Identification of Proposed Action: The action being considered by the Commission is the granting of exemptions from the inventory, record keeping and reporting requirements of 10 CFR 30.51, 40.61, 70.51(d), and 70.53 for core special nuclear, source and byproduct materials. Specifically, 10 CFR 30.51 and 40.61 require the maintenance of records showing the receipt, transfer and disposal of source or byproduct material. 10 CFR 70.51(d) specifies the requirements for the periodic conduct of a physical inventory of all special nuclear material in a licensee's possession. 10 CFR 70.53 specifies the requirements for the periodic submittal of a Material Balance Report and a Physical Inventory Listing for special nuclear material (SNM).

The Need for the Action: Given the severely damaged condition of the TMI-2 core fuel, the dislocation of fuel material from its original location in the reactor pressure vessel, and the nonhomogeneity of the dislocated

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material which has settled into piles of rubble at the bottom of the containment vessel, the licensee is unable to determine the bulk quantity of material in the vessel or to obtain representative samples for determination of source, byproduct, and SNM content in compliance with the core accountability requirements of 10 CFR 30.51, 40.61, 70.51(d) and 70.53. Accordingly, some relief from the Commission's regulatory requirements related to core accountability is warranted.

Environmental Impacts of the Proposed Actions: The staff has evaluated the exemptions and concluded that, as the exemptions are related to record keeping and reporting requirements, there are no significant radiological or nonradiological impacts to the environment as a result of this action.

Alternate to this Action: Since we have concluded that there is no significant environmental impact associated with the exemptions, any alternatives will have either no significant environmental impact or greater environmental impact. Alternatives to the exemptions would not reduce present environmental impacts of plant operations and would result in the application of overly restrictive regulatory requirements when considering the unique conditions of TMI-2.

Agencies and Persons Consulted: The NRC staff reviewed the licensee's request and did not consult other agencies or persons.




Alternate Use of Resources: This action does not involve the use of resources not previously considered in connection with the Final Programmatic Impact Statement for TMI-2 dated March 1981.

Finding of No Significant Impact: The Commission has determined not to prepare an environmental impact statement for the subject Exemption. Based upon the foregoing environmental assessment, we conclude that this action will not have a significant effect on the quality of the human environment.

For further details with respect to this action see; (1) Letter from F. R. Standerfer, GPUNC, to B. J. Snyder, USNRC, Core Accountability Exemption Requests, dated April 18, 1985.

The above documents are available for inspection at the Commission's Local Public Document Room, 1717 H Street, N.W., Washington, DC, and at the Commission's Local Public Document Room at the State Library of Pennsylvania, Government Publications Section, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

FOR THE NUCLEAR REGULATORY COMMISSION

  
Bernard J. Snyder, Program Director  
Three Mile Island Program Office  
Office of Nuclear Reactor Regulation

ENCLOSURE 3



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555

October 17, 1985

Docket No. 50-320

Docketing and Service Section  
Office of the Secretary of the Commission

SUBJECT: Three Mile Island Unit 2  
Approval of Exemption from 10 CFR 30.51, 40.61, 70.51(d),  
and 70.53

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- Notice of Proposed Issuance of Amendment to Facility Operating License.
- Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- Notice of Availability of NRC Draft/Final Environmental Statement.
- Notice of Limited Work Authorization.
- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).
- Notice of Issuance of Facility Operating License(s) or Amendment(s).
- Other: Exemption

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*Bernard J. Snyder*  
Bernard J. Snyder, Program Director  
Office of Nuclear Reactor Regulation

Enclosure:  
As Stated



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20555

September 16, 1985

Docket No. 50-320

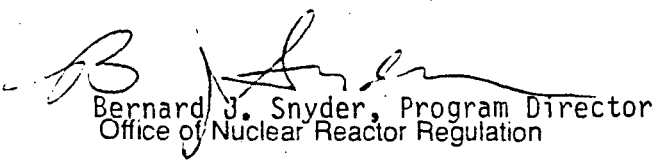
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Office of the Secretary of the Commission

**SUBJECT: Three Mile Island Unit 2  
Environmental Assessment and Notice of Finding of  
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- Notice of Availability of Safety Evaluation Report.
- Notice of Issuance of Construction Permit(s).
- Notice of Issuance of Facility Operating License(s) or Amendment(s).
- Other: Environmental Assessment and Notice of Finding of No  
Significant Environmental Impact

Enclosure:  
As Stated

  
Bernard J. Snyder, Program Director  
Office of Nuclear Reactor Regulation

Buckley, John | Friday, February 10, 2012

(12)

Document:  **Discusses status of review of NRC Technical Plan TPO/TMI-035 for core accountability. Determining collected weight by estimating volumetric quantities unacceptable. W/svc list. Record copy.** (Version 1.0, Released)

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Docket No. 50-320

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M-town Office

Mr. B. K. Kanga, Director  
Three Mile Island Unit 2  
GPU Nuclear Corporation  
P.O. Box 480  
Middletown, PA 17057

Dear Mr. Kanga:

Subject: TMI-2 Core Accountability

- References: (1) Letter, B. Kanga (GPU) to B. Snyder (NRC), TMI-2 Core Accountability, 4410-B3-L-0030, March 15, 1983.
- (2) Letter, B. Snyder (NRC) to J. Barton (GPU), TMI-2 Fuel Accountability, June 23, 1982.

The purpose of this letter is to apprise you of the status of our review of your Technical Plan (TPO/TMI-035) for TMI-2 core accountability. Your Technical Plan was forwarded by Reference (1) for our review. Upon receipt, we reviewed your plan to determine if the plan satisfied the regulatory guidelines for TMI-2 core accountability during defueling of the damaged core. Regulatory guidelines for TMI-2 core accountability were provided to GPU by letter dated June 23, 1982 (Reference 2). In the course of our review, several meetings have been held between members of our respective staffs (R. Weller and P. Grant of the TMIPO and R. Skillman, J. Byrne and J. Larson of GPU) to discuss the details of the planned defueling effort and the corresponding means to account for collected fuel. In these meetings, we have advised your staff about the aspects of your core accountability plan which are not in conformance with the regulatory guidance provided in Reference 2. Our guidance states that a net weight of each shipment should be obtained for that material that cannot be removed as a unit (e.g., a whole or partial fuel assembly). There is no provision in your plan for obtaining "net weights" for those materials (e.g., debris both within and outside the reactor pressure vessel) that will not be removed as either intact or partial assemblies. For debris and other dispersed material, the emphasis in your plan is on estimating "volumetric quantity" vice weighing to determine a net weight. We do not consider estimating volumetric quantities or other indirect means of determining a collected weight to be satisfactory for adequate core accountability. We are not suggesting that weighing need be performed in the reactor building or that any additional work need be conducted in high radiation areas. Net weights using commercially available equipment (e.g., load cell) can easily be obtained, for example, in the benign environment of the fuel handling building.

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NAME	RAWeller;hg	BSnyder	<b>8404206115XA</b>			
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Mr. D. K. Kanga

-2-

I bring this matter to your attention now to give you ample time to have a satisfactory core accountability plan in place prior to the initiation of defueling.

Sincerely,

Original signed by  
B. J. Snyder

Bernard J. Snyder, Program Director  
Three Mile Island Program Office  
Office of Nuclear Reactor Regulation

cc: J. Barton  
J. Byrne  
J. Larson  
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431 Park Avenue  
King of Prussia, PA 19406

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Chevy Chase, MD. 20015

Dr. Oscar H. Paris  
Administrative Judge  
Atomic Safety and Licensing  
Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Dr. Frederick H. Shon  
Administrative Judge  
Atomic Safety and Licensing  
Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Karin W. Carter  
Assistant Attorney General  
505 Executive House  
P.O. Box 2357  
Harrisburg, PA 17120

Dr. Judith H. Johnsrud  
Environmental Coalition on  
Nuclear Power  
433 Orlando Ave.  
State College, PA 16801

George F. Trowbridge, Esq.  
Shaw, Pittman, Potts and  
Trowbridge  
1800 M. St., NW  
Washington, D.C. 20036

Atomic Safety and Licensing Board Panel  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

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Washington, D.C. 20555

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Washington, D.C. 20555

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Dauphin County Commissioner  
P.O. Box 1295  
Harrisburg, PA 17108-1295

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Dauphin County Board of Commissioners  
Dauphin County Courthouse  
Front and Market Streets  
Harrisburg, PA 17101

Dauphin County Office of Emergency  
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Curtis Building (Sixth Floor)  
6th & Walnut Streets  
Philadelphia, PA 19106

Thomas M. Gerusky, Director  
Bureau of Radiation Protection  
Department of Environmental Resources  
P.O. Box 2063  
Harrisburg, PA 17120

David Hess  
Office of Environmental Planning  
Department of Environmental Resources  
P.O. Box 2063  
Harrisburg, PA 17120

Willis Bixby, Site Manager  
U.S. Department of Energy  
P.O. Box 88  
Middletown, PA 17057-0311

David J. McGoff  
Division of Three Mile Island Programs  
NE-23  
U.S. Department of Energy  
Washington, D.C. 20545

William Loehste  
104 Devey Laboratory  
Pennsylvania State University  
University Park, PA 16802

Randy Myers, Editorial  
The Patriot  
812 Market St.  
Harrisburg, PA 17105

Robert B. Borsum  
Babcock & Wilcox  
Nuclear Power Generation Division  
Suite 220  
7910 Woodmount Ave.  
Bethesda, MD. 20814

Michael Churchhill, Esq.  
PILCOP  
131 Walnut St., Suite 1632  
Philadelphia, PA 19107

Linda W. Little  
5000 Hermitage DR.  
Raleigh, NC 27612

Marvin I. Lewis  
6504 Bradford Terrace  
Philadelphia, PA 19149

Jane Lee  
183 Valley Rd.  
Ebers, PA 17319

J.B. Liberman, Esquire  
Berlack, Israel, Liberman  
26 Broadway  
New York, NY 10004

Walter W. Cohen, Consumer Advocate  
Department of Justice  
Strawberry Square, 14th Floor  
Harrisburg, PA 17127

Edward O. Swartz  
Board of Supervisors  
Londonderry Township  
RFD #1 Geyers Church Rd.  
Middletown, PA 17057

Robert L. Knupp, Esquire  
Assistant Solicitor  
Knupp and Andrews  
P.O. Box P  
407 N. Front St.  
Harrisburg, PA 17108

John Levin, Esquire  
Pennsylvania Public Utilities Comm.  
P.O. Box 3265  
Harrisburg, PA 17120

Honorable Mark Cohen  
912 E-E Main Capital Building  
Harrisburg, PA 17120

Mr. Edwin Kintner  
Executive Vice President  
General Public Utilities Nuclear Corp.  
100 Interpace Parkway  
Parsippany, NJ 07054

Blackley, John | Friday, February 10, 2012

Document:  **"TMI-2 Post-Defueling Survey Rept for Reactor Bldg Miscellaneous Components."** (Version 1.0, Released)

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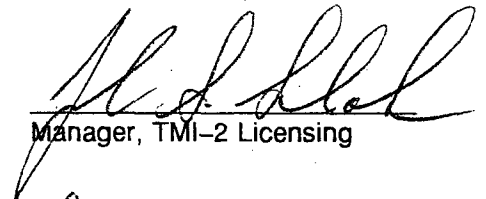
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TMI-2 POST-DEFUELING SURVEY REPORT

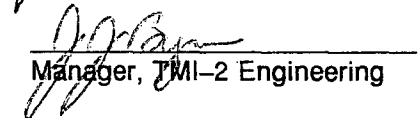
FOR

REACTOR BUILDING MISCELLANEOUS COMPONENTS

Approved:

  
\_\_\_\_\_  
Manager, TMI-2 Licensing

Approved:

  
\_\_\_\_\_  
Manager, TMI-2 Engineering

TMI-2 INTERNAL REVIEWS

AUTHORS:

  
F. W. DEININGER

RESPONSIBLE TECHNICAL REVIEWER:

  
M. H. HAGHIGHI

MANAGER, FUEL MEASUREMENTS & ANALYSIS:

  
C. H. DISTENFELD

PDSR COMMITTEE CHAIRMAN:

  
R. E. ROGAN

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## SUMMARY

The estimate of record of the amount of uranium dioxide ( $UO_2$ ) remaining in the RB Miscellaneous Components included in this Post Defueling Survey Report (PDSR) was 64.0 kg with a range from 31.6 to 85.4 kg, distributed approximately as follows:

Incore Guide Tubes (in "A" D-Ring)	21.0 kg $UO_2$
Fuel Transfer Canal/Transfer Tubes	18.9 kg $UO_2$
Upper Endfittings	5.9 kg $UO_2$
Core Flood System	4.9 kg $UO_2$
Drain Line from Tool Decon Facility	4.4 kg $UO_2$
Temporary RV Filter System	4.4 kg $UO_2$
Defueling Water Cleanup System	3.7 kg $UO_2$
Other Components	<u>0.8 kg <math>UO_2</math></u>
Total 64.0 kg $UO_2$	

The above summary table shows that 62% of the  $UO_2$  remaining in the RB Miscellaneous Components was located in two (2) components, the Incore Guide Tubes and the Fuel Transfer Canal/Transfer Tubes. The 21.0 kg  $UO_2$  in the Incore Guide Tubes (IGT) stored in the 'A' D-Ring was determined to be tightly compacted inside the 1.6 centimeter ID holes in the upper sections of the IGTs which was not accessible to the brushing tools. A small portion of the 18.9 kg  $UO_2$  remaining in the FTC/Transfer Tubes was fuel debris that was added during drain down of the Reactor Vessel.

The miscellaneous components were measured for fuel content using gross gamma exposure rates, gamma spectroscopy, neutron interrogation and direct sampling and analysis. These methods are described in Section 3.0 and highlighted for each component in Table 1. Several miscellaneous components not addressed in this PDSR are discussed in other PDSRs as listed in Section 1.0, Introduction.

The term "uncertainty" in this Post Defueling Survey Report was used to represent the estimated error of each "estimate of record" and was taken as one sigma. The overall uncertainty for the total residual  $UO_2$  remaining in the RB Miscellaneous Components is expressed as a range from 31.6 kg to 85.4 kg  $UO_2$  because of the asymmetrical value for the Fuel Transfer Canal/Transfer Tubes.

TMI-2 Post-Defueling Survey Report  
for  
The Reactor Building Miscellaneous Components

1.0 INTRODUCTION

This report presents the analysis of the amount of fuel (UO<sub>2</sub>) remaining in the Reactor Building for miscellaneous components only. The content of this analysis addresses the fuel remaining in the following components/systems.

- a. Reactor Coolant Drain Tank
- b. Fuel Transfer Canal/Transfer Tubes
- c. Core Flood System
- d. Upper Endfittings
- e. Tool Decon Facility (TDF)
- f. Drain Line from TDF
- g. Defueling Water Cleanup System
- h. Temporary RV Filter System
- i. Incore Guide Tubes
- j. Defueling Tools

The remaining miscellaneous components in the Reactor Building are addressed in other Post-Defueling Survey Reports as listed below:

<u>Miscellaneous Component</u>	<u>Addressed in PDSR</u>
Rv Head	RV Head Assembly
Plenum	Plenum Assembly
RB Basement	RB Basement
Letdown Coolers	Letdown Cooler Room



This report is one in a series of reports prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 1). All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated. Other segments of the RB are treated in separate Post-Defueling Survey Reports.

Section 2, "Background," describes the physical attributes of the individual components/systems and their relationship to the accident and subsequent cleanup activities. The boundaries of these components/systems with respect to this PDSR are also presented in this section.

Section 3, "Measurement Methods," describes the measurement methods utilized to assay the residual fuel ( $UO_2$ ) in each of the components/systems. The majority of components/systems were assayed using passive nondestructive assay methods (i.e., gamma spectroscopy, gamma dose rate, or neutron interrogation).

Section 4, "Analysis," explains the methodology for arriving at the estimate of record of fuel ( $UO_2$ ) in the RB using multiple measurements and sample analysis data and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of SNM remaining in the RB and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analyses performed.

## 2.0 BACKGROUND

The March 1979 TMI-2 accident resulted in significant damage to the reactor core and subsequent release of fuel particles and fission products into the Reactor Coolant System (RCS) and other closely interconnected systems/components. The damaged core consisted of loose fuel pellets, resolidified fuel, structural metal components, loose rubble, and partial fuel assemblies. Therefore, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

During the accident, core debris was transported to the Reactor Building (RB) as a result of core degradation and coolant flow from the reactor vessel (RV) through the pilot operated relief valve (PORV) and the RCS Makeup and Purification (MU&P) System. Approximately 10 kg of fuel ( $UO_2$ ) was relocated to the RB sump and other RB locations during the accident sequence (Reference 2). Subsequent to the accident, fuel was relocated to the RB as a result of several cleanup operations including: transfer to and storage of structural RV components in the 'A' core flood tank (CFT) and 'A' D-ring; storage of upper endfittings; flushing of defueling tools; and transfer of the defueling canisters into the fuel transfer canal (FTC). Even though fuel was relocated to the RB during cleanup operations, RB residual fuel conditions were maintained significantly below the safe fuel mass limit (SFML), Reference 3. Further, a significant cleanup effort was undertaken with the primary purpose of reducing exposure rates but which also resulted in the removal of additional core debris.

### 2.1 Reactor Coolant Drain Tank

The reactor coolant drain tank (RCDT) was designed to absorb blowdown from the pressurizer relief valves and receive leakage from the RC pump seals and RC valves. The discharge from the RCDT was

designed to flow through a cooler to the reactor coolant bleed tanks (RCBT). The RCDT normally had a nitrogen cover gas and was vented to the RCBTs. The tank was protected by a 150 psig relief valve and a 200 psig rupture disk.

During the accident, reactor coolant was discharged from the RCS through the PORV to the RCDT which was located in the basement of the Reactor Building. The RCDT contained two (2) safety components: a relief valve which discharged to the RB sump and a rupture disk which discharged to the RB floor adjacent to the RCDT cubicle. Both safety devices were believed to have performed their respective safety functions. The rupture disk was subsequently found in an open or ruptured condition, as expected. During the accident, the operators realized that the PORV was not closed and they manually closed the pressurizer block valve. However, the block valve had to be cycled repeatedly to maintain system pressure. This cycling of the block valve permitted the transport of fission products, noble gases and small quantities of fuel through the pressurizer, PORV and block valve into the RCDT and subsequently into the reactor building through the rupture disk discharge.

## 2.2 Fuel Transfer Canal/Transfer Tubes

A small amount of fuel resides at the bottom of the fuel transfer canal (FTC), having been transported from the RV to the FTC as debris adherent to the outside of the fuel bearing canisters. Additional fuel was deposited in the FTC as a result of the placement and storage of the upper plenum assembly in the deep end of the FTC. During the accident, fuel particles were transported to the plenum when large amounts of reactor coolant flow, steam, and hydrogen passed through it. Fuel was deposited in the form of sediment and surface films on the plenum surfaces. Prior to its

removal from the RV, the plenum was flushed to remove loose surface debris. The fuel content of the plenum was reported in Reference 4 and will not be included here. Additional fuel deposits are contained with the endfittings stored in a temporary storage container (TSC #3) which is stored in the FTC.

During drain down of the RV and connecting systems/components, water will be transferred to the deep end of the FTC. The FTC will act as a settling volume to collect any fuel fines removed during drain down. The RV will be pumped down in multiple phases. The first phase will be to drain the water from the RV using a jet pump. The jet pump's suction will be located in the area where the H-8 boat sample was removed during the RV sampling program. Then the RV will be refilled using water from the RCBTs. During drain down and refilling, the RV will be measured for remaining fuel ( $UO_2$ ). After the fuel measurements have been completed, draining of the connected systems/components will begin. The RV will be drained to approximately half-full and then the drain pumps in the steam generators will be turned on so that they are draining to the RV. During the draindown of the connected systems/components, the RV water level will be maintained well below the nozzle level (elevation 311') to prevent water from flowing back into the cold legs via the nozzles.

The pumping rate in and out of the RV will be approximately 10 gpm. Once both steam generators have been drained, the RV will be drained. All of the water removed from the steam generators and RV will be pumped directly to the FTC. During the drain down, it will be necessary to continuously remove water from the FTC to prevent the FTC from overflowing. The water removed from the FTC will be pumped to a RCBT. The drain line from the RV to the FTC will be located at least fifteen (15) feet away from the suction line to the

RCBTs to prevent carry over of fuel fines.

### 2.3 Core Flood System

Two (2) core flood tanks (CFT) were installed as part of the original plant design as a passive low pressure injection system in the event of a loss of coolant accident. They were designed to inject borated water directly into the RV when the RCS pressure dropped below a predetermined level. Approximately eight (8) hours into the accident, the core flood tanks injected some water into the RV. However, twelve (12) minutes later, a high level alarm indicated that the core flood tanks were receiving water from the RCS even though a check valve was installed to prevent such an occurrence (Reference 2). Fuel fines could have been transported into the core flood system by this backflow. A more likely means of fuel transport took place after the initial core damage. The water level in the RV is believed to have dropped below the core flood nozzles on two (2) separate occasions, thereby draining the horizontal lines. It is likely that fuel was transported into these lines by the fluid expansions resulting from the high energy releases inside the RV or by the refilling operations.

During the lower core support assembly (LCSA) defueling, the top of the 'A' CFT was removed and the tank was used for storage of LCSA components. Additionally, the piping from the 'A' CFT to the RV was cut and flanged which prevented the possibility of further fuel transport. Storage of the LCSA components in the 'A' CFT permitted continuous progress in the RV defueling activities. Prior to removal from the RV, the LCSA segments were video inspected to ensure that no visible fuel was present. Sample sections of each plate were measured by gamma spectroscopy and/or alpha measurements to determine the quantity of adherent residual fuel.

## 2.4 Upper Endfittings

In September 1986, the core bore machine was installed in the RV to break up the large resolidified mass so that it could be removed and placed in fuel canisters. Loose upper endfittings that would interfere with drilling operations were removed from the surface of the debris bed. In some cases, several endfittings had been fused together and were too large to fit into the fuel canisters; they were placed in shielded drums filled with borated water and stored at elevation 347' in the reactor building near the RV head stand.

Currently, there are approximately seventeen (17) upper endfittings stored in six (6) containers. The maximum number of endfittings in a single container is four (4) (See Section 4.4).

## 2.5 Tool Decontamination Facility

The tool decontamination facility (TDF) was located on the 347' elevation of the RB and consisted of two (2) 12 foot square enclosures connected by an 8 foot square anteroom. One enclosure was used for high pressure/high temperature flushing of defueling tools; the other enclosure was used for cutting up the defueling tools. Because of the extensive use of this facility, residual fuel deposits remain in the form of surface contamination.

## 2.6 RB Drain Line From TDF

The RB basement boundary was taken to include all space below the 305' elevation with one exception. The exception is the RB drain line that was used to transfer defueling tool decontamination wash water to the basement. A separate Post Defueling Survey Report has

been submitted for the RB basement and determined the fuel Estimate of Record for the RB basement at 1.3 kg UO<sub>2</sub>. The discharge path from the tool decontamination facility located on the 347' elevation of the RB is from the decon sink to the floor drain located within the decon facility. The discharge piping from the floor drain passes through the 347' elevation floor, turns nearly horizontal for approximately ten (10) feet and then is essentially vertical for approximately fifty-five (55) feet to a long horizontal flooded drain line under the 282' elevation basement floor (Reference 5). More than a dozen basement floor drains empty into the line. The residual fuel exists in the line as small pebbles as opposed to the colloidal particles that were transported to the RB sump (Reference 2). It is noteworthy that the defueling tools were flushed in the RV prior to being removed to minimize fuel transport.

## 2.7 Defueling Water Cleanup System

The defueling water cleanup system (DWCS) located in the RB was composed of three (3) parts: the interconnecting hoses, the manifold assembly, and the RV cleanup pumps located in the south end of the FTC canal. All of the filters have been removed and shipped off-site. The interconnecting hoses consisted of two (2) similar assemblies labeled train 'A' and 'B'. In aggregate the trains of reinforced rubber hose consisted of approximately twenty-eight hundred (2800) feet of nominal two (2)-inch diameter hose; fourteen hundred (1400) feet of nominal three (3)-inch diameter hose; and fourteen hundred (1400) feet of nominal four (4)-inch diameter hose (Reference 6). The fuel content was determined by direct gamma spectrometry of representative hose samples and extensive directional detector gross gamma scans for a larger sample.

The manifold assembly was about twenty-one (21) feet long and was

located along the north end of the FTC. It was composed of a complex of hard piping, valves and other components that interfaced between the RV and filter canisters to provide the cleaning action. Since the filter canisters have been removed, there is no allowance for fuel residual to the filters. Therefore, the entire manifold assembly was in the field of view of a directional detector and shielded only by individual pipes and hoses (Reference 6).

The third section of the DWCS was two (2) RV cleanup pumps. The two pumps contained suction lines from the RV and discharge lines to the manifold assembly. The pumps were located under water in the former fuel storage pit near the southeastern corner of the FTC. The pumps were emersion type, designed to fit inside vertical 10 inch steel casings. The suction side of the casings were terminated by 10 inch caps that were continued by schedule 40 pipe sections welded to the cap centerline. The short downgoing 4 inch schedule 40 pipe section became horizontal for 18 inches and then extended vertically up and out of the fuel storage pit. The entire DWC system was utilized over several years while defueling was in progress (Reference 5).

## 2.8 Temporary RV Filtration System

The function of the Temporary RV Filtration System (TRVFS) was to restore and maintain the visibility in the RV to allow continuation of the early defueling program. Investigation had lead to the discovery of micro-organism growth in the reactor coolant. Operating experience revealed that these micro-organisms were capable of plugging the filters prior to the collection of any significant quantity of core debris. The TRVFS system was operated as a temporary filter system while GPU developed a permanent program to control this phenomenon.



The TRVF System consisted of a pump, 1 1/2-inch diameter suction and discharge hoses, isolation valves, fittings, and filter assemblies. The filter assemblies were commercial type diatomaceous earth-filled swimming pool filters. The suction depth was controlled by fixed lengths of solid piping. The TRVFS was operated only when operations personnel were on the defueling platform. The TRVFS took suction from the IIF and/or the RV and returned the filtered water to the IIF. The initial flow rate through the filter was approximately 100 gpm. When the pressure drop across the filter reached a predetermined level (10 psi), the pump could no longer provide significant flow (Reference 7). Consequently, the filter was "backbumped" or cleaned. Backbumping was accomplished by stopping the flow and flexing the filter media which caused the filtered material and diatomaceous earth to fall to the bottom of the filter housing. The TRVFS was eventually replaced by the Defueling Water Cleanup System which is described in Section 2.7. The two (2) highly radioactive swimming pool filters are currently stored in the RB on elevation 305'.

## 2.9 Incore Nozzles

Sections of the flow distributor removed from the RV, which contained incore guide tubes, were too large to be placed in the 'A' CFT. Thirteen (13) such sections, containing 33 incore guide tubes, were stored in the 'A' D-ring. These sections were bagged and suspended in the 'A' D-ring in order to prevent interference with the RV defueling efforts. Each section was brushed and flushed prior to removal from the RV.

The thirteen (13) sections of the flow distributor stored in the 'A' D-ring, with incore guide tubes attached, generally consist of a one (1) to two (2) foot rectangularly shaped piece of the flow

distributor with up to four attached incore guide tubes of up to forty-seven (47) inches in length. The majority of the residual fuel associated with these sections was determined to be tightly compacted inside the 5/8-inch inside diameter holes in the upper section of the incore guide tubes and not accessible to the brush tools.

## 2.10 Defueling Tools

Typical defueling tools consisted of stainless steel pipe sections two (2) to four (4) inches in diameter and thirty (30) to forty (40) feet long with an end effector attached. For the purpose of this assessment, the defueling tools were modeled using Microshield (Reference 8). The inside surfaces of the pipe were assumed to be open to the internal environment of the reactor vessel and contain fuel debris. Routine practice required the working end of each tool removed from the RV be isolated in a plastic bag. This contamination control technique also prevented the spread of fuel bearing material to the tool racks or to surfaces below the tool storage areas. The radiation environment in proximity to the rack containing the most contaminated tools was not distinguishable from normal values above the 'B' D-ring.

### 3.0 MEASUREMENT METHODS

Post-defueling SNM measurement in the TMI-2 Reactor Building was a complex task. Several different methods were used to locate and quantify residual fuel. These methods included direct measurement by instrumentation, visual inspection, and sample collection and analysis. The methods selected were influenced by many factors including accessibility, configuration of component/area to be assayed, area radiation dose rates, measurement uncertainties, and equipment sensitivity.

The following includes a discussion of the various methods and the factors that influenced their selection. Five (5) general methods were used for fuel detection: detection of gamma rays, neutrons, alpha particles, sample and analysis, and visual inspection. Each detection method included a number of specific techniques that are described below.

#### 3.1 Gamma Dose Rate

Gamma detection for fuel measurement included the use of the gross gamma exposure rates and gamma spectroscopy techniques. Gross gamma directional surveys were performed on areas, piping and components when exposure rates and accessibility permitted. The measurement system utilized a portable directional gamma survey instrument consisting of a directional probe and readout device. TMI-2 core debris contained fission and activation products which produced gamma-ray exposures. Exposure rate values were calculated based on the isotopic distribution of sample data. Using the reactor core debris sampling data, a correlation was developed between the gross gamma-ray output and the quantity of fuel present (Reference 9). Computer modeling was used to predict exposure rates for a given geometry of fuel for a particular location. Field measurements were performed for specific locations and compared with the computer

modeled output. This comparison was done on a point by point basis. Scaling the models with the measured exposure rates yielded an estimate of residual fuel. Three (3) areas/components (decon facility, DWCS-manifold and fuel transfer canal/tubes) were analyzed using gross gamma directional surveys.

### 3.2 Gamma-Ray Spectroscopy

Gamma-ray spectroscopy was used to quantify the amount of a particular radioactive isotope present by measuring the characteristic gamma radiation emitted. Typically, the emitted gamma radiation was detected by a high-purity germanium (HPGe) detector. The detected radiation impulses were converted to an electrical signal which, when processed by an analyzer, identified the relative energy of the originally emitted gamma-ray radiation. Calibrated gamma-ray spectroscopy systems were used at TMI-2 to measure the quantity of Ce-144(Pr-144) and/or Eu-154 present in discrete locations or on components. The quantity of cerium or europium present was converted to the quantity of fuel present based upon the calculated ratios (Reference 10) and the actual measurements of the cerium/fuel and europium/fuel ratios.

The HPGe detector measurements were performed on the defueling tools, DWCS hoses, LCSA sections stored in the 'A' Core Flood Tank and incore guide tubes stored in the 'A' D-ring. The HPGe detectors are approximately 1 3/4" diameter x 1 3/4" long while the NaI detectors are approximately 1/2" diameter x 3/4" long. HPGe detectors have the advantage of much better energy resolution capability due to their relative size, than sodium iodide (NaI) detectors. In addition, they are much more sensitive to ambient gamma radiation levels. However, HPGe detectors require liquid nitrogen cooling to operate. HPGe detector measurements were

performed to identify both Ce-144/Pr-144 (2.19 MeV gamma radiation) and Eu-154 (1.27 MeV gamma radiation). The Ce-144/Pr-144-and Eu-154-to-fuel (UO<sub>2</sub>) ratios were 152.5 and 42.6  $\mu\text{Ci g}^{-1}$ , respectively as of 8/1/87 (Reference 10).

The HPGe detector was housed in a 2-inch thick cylindrical shell lead shield. The detector was connected to a preamplifier, amplifier, and multi-channel analyzer (MCA) as described in Reference 11. The MCA and input-output device were at a remote location; as much as 100 feet of signal cable was connected between the preamplifier and spectroscopy system. The detector system was source checked periodically to ensure proper operation using standard Ce-144 and Eu-154 sources. After the calibration data had been collected, the standard sources were removed and the acquisition of data was performed.

### 3.3 Neutron Interrogation

Neutron emission from the induced fission reactions is directly proportional to fuel quantity. This active neutron assay method interrogates fuel with lower energy neutrons and detects induced higher energy fission neutrons.

Active neutron interrogation is more sensitive than some passive methods for quantifying small deposits of fuel. At TMI-2, an antimony-beryllium (Sb-Be) photoneutron interrogation method used an antimony photon source of 1.7 MeV gamma-rays to produce low-energy (approximately 0.024 MeV) interrogating neutrons via the Be ( $\gamma, n$ ) reaction. These interrogating neutrons which are moderated by the beryllium ring (and polyethylene surrounding the Be-ring) impinged upon the fuel and induced fission reactions in the fissile material contained in the fuel. Some of the fission neutrons emitted from

the surrounding fuel were detected by a He-4 fast neutron proportional counter. The He-4 neutron counter system differentiated the higher-energy induced fission neutrons from the lower-energy photoneutron source and gamma-rays on the basis of the pulse height signal. With proper shielding, this system can operate effectively and efficiently in a substantial radiation field.

### 3.4 Direct Sampling and Analysis

Two types of samples were used at TMI-2 for residual fuel determinations: core debris and RCS components. Samples of core debris from fixed locations were analyzed to determine fuel and radionuclide content. Samples were analyzed by gamma spectroscopy, alpha counting, and chemical/physical techniques. Estimates of the total debris volume of concern were developed using visual aids and radiological models were incorporated into the analytical results to derive specific fuel quantities. Samples of RCS components were used to estimate the density of fuel fixed in surface films. Representative samples of various core debris deposits were extrapolated to represent deposits from the reactor vessel. The distant material has greater potential gamma activity per unit fuel weight. Because of the uncertainty in the sample representation, it was preferable to use sampling techniques in conjunction with other methods that measure fuel directly.

Two (2) areas/components (temporary reactor vessel filter system and the RC drain tank) were assayed using direct sampling and analysis.

#### 4.0 ANALYSIS

The Reactor Building (RB) miscellaneous areas were measured for fuel ( $UO_2$ ) by performing a series of non-destructive measurements described in Section 3.0, Measurement Methods. The estimate of record for each component/system was determined by performing gamma detection measurements or neutron interrogation measurements and/or engineering analyses. Engineering analyses were performed in areas where the total fuel quantity was believed to be insignificant. An insignificant quantity was defined as about 7 kg of  $UO_2$  (Reference 1) or approximately 8% of the maximum safe fuel mass (Reference 3) for fuel removed from or remaining in the reactor vessel.

All assessment measurements were performed using a direct gamma reading or counting technique and are supported by formal engineering calculations containing data sheets, equipment calibration information and essential supporting information as well as data reduction. It is assumed, unless otherwise stated, that fuel estimates based on existing references are correct within the uncertainties stated. The result of these fuel assessments was to provide an "estimate of record" of residual fuel ( $UO_2$ ) for the miscellaneous areas in the Reactor Building.

Two (2) major components contained 62% of the residual fuel in the RB Miscellaneous Components. They are the Fuel Transfer Canal/Transfer Tubes and the Incore Guide Tubes stored in the 'A' D-Ring. Other miscellaneous components contained quantities of fuel as listed in Table 2. The analysis for each component is discussed individually below.

The term "uncertainty" in this Post-Defueling Survey Report is used to represent the uncertainty of each "estimate of record" and is taken to be one sigma. The uncertainty was determined by the project engineer for each individual measurement and appropriate engineering calculations. The uncertainty value was derived by taking into consideration uncertainties in

mathematical modeling, measurement equipment, counting, standard values (fissionable material), randomness, and uncertainty in background levels. The range of uncertainty for the total residual  $UO_2$  remaining in the RB Miscellaneous Components was determined using the asymmetrical term for the fuel transfer canal/transfer tubes and combining its range algebraically with the fuel quantity (45.1 kg  $UO_2 \pm 32\%$ ) estimated for the symmetrical terms. The parts leading to the overall uncertainty were considered for possible covariant uncertainty. Each ingredient is considered to be independent.

This results in an estimate of record of 64.0 kg  $UO_2$  for all the RB Miscellaneous Components as shown in Table 2. After evaporation of the accident generated water stored in the components, the potential for fuel mobility will be greatly reduced because only small batches of water will accumulate and be processed during the post defueling storage period.

#### 4.1 Reactor Coolant Drain Tank

The west end of the Reactor Coolant Drain Tank (RCDT) was visually inspected by lowering a camera from the 305' elevation in the RB. A four (4)-inch diameter hole was drilled in the concrete floor (elevation 305') and in the elbow of the eighteen (18)-inch vent line directly below. A small cylindrical underwater black and white camera was lowered into the RCDT. Because of limitations inherent in the equipment used to perform the video inspection, only the west end of the tank was examined. A sediment sampler was also used to collect samples of liquid and particulate matter from the inside bottom surface of the RCDT, directly beneath the rupture disk and vertical section of the vent line. Additional details are discussed in Reference 12.

Video surveys of the inside of the RCDT showed that the layer of



debris was quite thin in the area beneath the rupture disk. For the purpose of this PDSR, the debris on the bottom surface of the tank was assumed to be 0.16 cm thick and to uniformly cover one-eighth of the inner surface of the tank from end to end. The area of the deposited debris was about  $52,000 \text{ cm}^2$ , and its total volume was approximately  $8320 \text{ cm}^3$ . Based on the elemental analysis results for the RCDT solids sample presented in Reference 13 (the elements listed collectively account for about 94% of the sample weight), the density of the debris was about  $6.2 \text{ g/cm}^3$ . If we assume that one-half of the total debris volume was water for a packing fraction of 0.5, then the total mass of debris in the RCDT was about 26 kg ( $8.3 \times 10^3 \text{ cm}^3 \times 6.2 \text{ g/cm}^3 \div 2$ ). See Reference 13 for details of results.

The microprobe analyses performed on the RCDT samples did not definitely identify uranium, although cerium, which is generally associated with uranium, was detected in each sample using gamma-ray spectroscopy measurement techniques. Neutron activation analyses subsequently performed on the RCDT solids quantified the uranium in the samples. The concentration of uranium measured through neutron activation analysis was 3.7 mg/g. The total fuel content of the RCDT was estimated as  $(26,000 \text{ gm}) (3.7 \text{ mg/g}) (1.1345 \text{ UO}_2/\text{U ratio}) = 0.1 \text{ kg UO}_2$ .

The uncertainty is dominated by two components. The lesser uncertainty is due to the radiochemical fuel analysis of the RCDT sample. EG&G Idaho and the Westinghouse Hanford Engineering Development Laboratory independently analyzed the sample. A reasonable uncertainty of 20% (Reference 14) was adopted for the fuel assay since specific error values were not published. The larger uncertainty is assigned to the determination of sediment volume lying on the bottom of the RCDT. Previously, a video estimate was performed on the upper tube sheet of the "B" Once Through Steam Generator,

Reference 15. Subsequently the work was reassessed by gross gamma methods, Reference 16. Generally the video method produced a result that was less accurate than the gross gamma method. Since the radiochemical and video uncertainties are independent, the overall uncertainty was propagated as the product of volume and concentration as the square root of the sum of the squares of the fractional errors. The overall uncertainty is 54%.

#### 4.2 Fuel Transfer Canal/Transfer Tubes

During the TMI-2 Cleanup Program, fuel canisters were transferred from the RV through the FTC/transfer tubes for temporary storage in the Fuel Storage Pool A in the Auxiliary Building. The residual fuel debris in the FTC came from six (6) different sources. A major source of fuel debris was the fuel canister transfer when they were lowered into the FTC and rotated from vertical to horizontal for transfer to the Spent Fuel Pool A (SFPA). Three hundred and one (301) fuel canisters were transferred through the FTC prior to measurement of the FTC (Reference 17). The term fuel canister is used in the all inclusive sense; i.e., "fuel", "knockout" and "filter" canisters. Underwater gross gamma measurements were performed in the FTC during August and September 1989. Also a video inspection was made before and during these measurements to determine the distribution patterns for the residual fuel debris. The video inspection disclosed that there were not any areas where a significant depth of debris could be seen and in fact many areas appeared relatively clean. The measured fuel ( $UO_2$ ) quantity of 12.2 kg used a Cs-137 activity to fuel ratio of 1800  $\mu Ci/g UO_2$ . Since the FTC measurement was performed in 1990, the new Cs-137 activity to fuel ratio value of 1355  $\mu Ci/g UO_2$  (Reference 9) was determined which changes the FTC fuel uncertainty to +24%, -100%. The result of this improved Cs-137 fuel activity is that the FTC initially is estimated to contain 12.2 kg  $UO_2$  +24%, -100%. The

basic measurement data and details of the data analyses are contained in Reference 18. Since the measured fuel quantity was the result of transferring three hundred and one (301) canisters, it was reasonable to assume that the increase in residual fuel can be prorated for the last forty-one (41) canisters (Reference 19) transferred after the measurements were taken. This additional fuel deposit was estimated by multiplying the last forty-one (41) canisters times the average fuel per canister transferred with the initial three hundred and one (301) canisters.

Average fuel/canisters =  $(12.2 \text{ kg}) \div (301) = 0.041 \text{ kg UO}_2/\text{canister}$   
Additional fuel deposited =  $(0.041 \text{ kg}) (41 \text{ canisters}) = 1.7 \text{ kg UO}_2$

Therefore, the total estimated fuel ( $\text{UO}_2$ ) in the FTC as a result of transferring canisters to SFPA was  $12.2 \text{ kg} + 1.7 \text{ kg} = 13.9 \text{ kg}$ , +24%, -100%.

Another contributor to the fuel ( $\text{UO}_2$ ) remaining in the FTC was the fuel debris remaining in the fuel transfer tubes which connect the FTC to the SFPA. The fuel transfer tubes were assumed to have fuel debris deposited in the lower sixty (60) degrees of arc of each tube because the tubes are basically horizontal. Reference 18 determined the residual fuel ( $\text{UO}_2$ ) deposited in the transfer tubes as  $0.5 \text{ kg}$ , +34%, -92%.

A third contributor to the fuel ( $\text{UO}_2$ ) remaining in the FTC was a storage container (designated TSC) containing three (3) endfittings from the original fuel assemblies. The fuel debris contents of this storage container is included in the total fuel estimate for the Upper Endfittings discussed in Section 4.4. Even though the container itself is stored in the deep end of the FTC, it seems appropriate to include its fuel estimate along with the other endfittings stored on

the 347' elevation in the Reactor Building.

A fourth contributor to the fuel ( $\text{UO}_2$ ) remaining in the FTC was the upper reactor Plenum Assembly. Based on the video inspections (which saw no debris) and the fact that the plenum was flushed prior to placing it in the FTC, the plenum was assumed to contain less than 0.1 kg of fuel ( $\text{UO}_2$ ) in the form of loose debris or rubble. Reference 20 describes the analysis utilized and assumptions made to estimate the quantity of fuel ( $\text{UO}_2$ ) remaining on the plenum assembly. The fuel remaining on the plenum assembly is not included in the total fuel ( $\text{UO}_2$ ) estimated for the FTC because it is addressed in a separate Post Defueling Survey Report (Reference 4).

A fifth contributor to the fuel ( $\text{UO}_2$ ) remaining in the FTC was attributed to approximately seven-hundred (700) feet of three (3) different sizes of rubber hose (Reference 21 and 22) that were installed between the DWCS manifold and the filter canisters installed in the deep end of the FTC. This seven-hundred (700) feet of hose was removed from the work platform and placed in the FTC. The DWCS Engineering Calculation, Reference 6, summarized the fuel concentration that was measured on sample lengths of hose surfaces exposed to unfiltered RV water. The average fuel mass ( $\text{UO}_2$ ) for several lengths of sample hoses was  $1.5\text{E}-5$  kg  $\pm 100\%$  for approximately 486 square centimeters of inner surface area. Applying this average value to the seven-hundred (700) feet of hose ( $376999$   $\text{cm}^2$  of inner surface) results in a total estimated fuel deposit of 0.01 kg  $\pm 100\%$ .

The sixth and final contributor to the fuel ( $\text{UO}_2$ ) remaining in the FTC is the fuel debris ( $\text{UO}_2$ ) being transferred to the FTC during drain down of the RV and connecting systems/components.

Following the completion of pick and place activities in the lower

head region of the RV, the lower head was vacuumed to minimize the relocation of core debris to other surfaces in the vessel and to improve the visibility in the RV. The in-vessel vacuum system, utilizing a knockout canister and filter canister in series, was used for this evolution. Following the NRC-sponsored Lower Head Sampling Program, a final cleanup of the lower head region was performed during March 1990, and a revised estimate was determined. There appeared to be a fine dusting of material distributed over the entire bottom head surface except for two patches where granular material was somewhat deeper (an average of 0.3 cm deep). Based on this information, it was estimated that the remaining fuel debris in the lower head region was 8.1 kg UO<sub>2</sub> (Reference 23).

In order to measure the RV for remaining fuel (UO<sub>2</sub>) it will be necessary to completely drain and refill the RV while making active and passive neutron measurements. It is estimated that most of the fuel debris transferred from the RV to the FTC during drain down will occur while the jet pump is draining the RV from level 8 (forging location) to the bottom of the lower head. Each time the RV is drained a quantity of core debris will be transferred to the FTC. It is necessary to perform this drain sequence at least once (1) during the RV fuel measurement program and again during the final drain down sequence.

Once the jet pump is installed in the lower head, the pump suction will be close enough to the bottom surface and the water velocity will be high enough to remove a quantity of the fine debris which is located near the pump intake port.

During the first drain down cycle, it is estimated that approximately 1.1 kg of UO<sub>2</sub> will be transferred from the RV to the FTC. This value, approximately 13% of the loose fuel on the lower head, was determined

using the volume of core debris existing within the range of flow velocities greater than 0.3 centimeters per second. This assumes that the pump picks up all particles uniformly. See Appendix A for definition of the area from which pump suction removes fuel debris. This is an area surrounding the jet pump approximately ten (10) inches distant from the intake port. It also assumes that the debris is initially 0.2 centimeters deep. Using an average debris density of  $4.4 \text{ g/cm}^3$  (Reference 24), and an average  $\text{UO}_2$  debris ratio of 0.6 (Reference 25), the total mass of  $\text{UO}_2$  transferred from the RV to the FTC is estimated as  $1.1 \text{ kg UO}_2 (25.4 \text{ cm} \times 25.4 \text{ cm} \times \pi \times 0.2 \text{ cm} \times 4.4 \text{ gm/cm}^3 \times 0.6)$ .

After the vessel has been completely drained the first time, it will be refilled using water from the RCBTs via the 'B' makeup line (MU-V-16B). During refilling, it will be necessary to stop at discrete locations to perform a series of fuel measurements. This filling operation will tend to wash most of the residual fuel out of the '1B' cold leg and nozzle. The '1B' cold leg and nozzle were estimated (see discussion below) to contain approximately 26 kg  $\text{UO}_2$ . In addition, the filling action will tend to slosh water through a section of the downcomer region which will tend to move small quantities of available loose, fine debris from the horizontal surfaces down to the lower head. The velocity of the water inside the Core Support Assembly during refilling is estimated to be less than 0.04 cm/sec (100 GPM) and therefore, very little of the remaining fuel debris will be relocated inside the Core Support Assembly. Prior to performing the neutron measurements in the RV, the estimated quantity of loose fine debris in the lower Downcomer/Core Support Assembly region was approximately 18.1 kg  $\text{UO}_2$  emerging from the thermal shield support blocks (15.2 kg) and core catchers (2.9 kg) - Reference 23. It is not unreasonable to assume due to localized higher water velocities that approximately 25% of the 18.1 kg or 4.5 kg  $\text{UO}_2$  of loose debris will be

redistributed during the first complete filling of the RV. This is the first time since the accident that the RV will be completely drained and refilled, and a small amount of sloshing in a limited area will take place during filling.

The '1B' cold leg and nozzle contain a significant quantity of fuel debris which could be relocated during refilling of the RV. The '1B' cold leg was estimated to contain 2 kg  $UO_2$  (Reference 26). The nozzle connected to the '1B' cold leg contains approximately 24 kg  $UO_2$  which was estimated based on its volume of fuel debris (13,000  $cm^3$  per Reference 27), the density of the fuel debris (3.2  $g/cm^3$  per Reference 26) and the  $UO_2$ -to-fuel debris ratio (58% per Reference 25). The product of the three (3) factors above is 24.1 kg  $UO_2$ . The fuel estimate for the '1B' cold leg and nozzle quantities is 26 kg  $UO_2$ . Combining the 26 kg in the '1B' cold leg and nozzle with the 4.5 kg from the thermal shield support blocks and core catchers (downcomer region), and the 7.0 kg remaining on the lower head, implies that approximately 38 kg  $UO_2$  of loose debris could be relocated into the lower head region during the RV refilling sequence.

After the neutron measurements have been completed, it may be necessary to drain the RV a second time. During this second draining, it is estimated that a small amount of the loose fine debris relocated during refilling of the RV could be pumped to the FTC. Since the RV drain pump sucks from an area equivalent to approximately 3% of the lower head area and the lower head is sloped towards the drain pump, it would be reasonable to assume that 5% (1.9 kg  $UO_2$ ) of the relocated fuel could be transferred to the FTC during the second drain down of the RV.

The next sequence will be to drain the OTSGs to the RV. It is assumed that a portion of the residual fuel in the OTSGs will be pumped to the

RV and consequently a portion of that fuel material will be transferred to the FTC. It will be necessary to partially drain the RV periodically while the OTSGs are being drained to the RV. During this sequential draining, it is assumed that loose fuel debris will be relocated within the RV. In addition, fuel fines coming from the OTSGs will settle out on horizontal surfaces in the RV and the lower head region.

The data collected during the Once Through Steam Generator (OTSG) measurement program was evaluated and fuel quantities in the 'A and B' OTSG lower heads and J-legs were estimated. This was accomplished by comparing the actual in situ exposure rates to calculated rates for known activity concentrations modeled with the Microshield Computer Program (Reference 8). The debris observed in both OTSGs was very low density material and was easily suspended while moving the video probe. This low density material was uniformly distributed except for a few areas which contained gravel like debris which corresponded to the highest exposure rates. The resulting fuel estimates prior to drain down for the 'A' OTSG lower head and J-legs were 0.29 kg and 0.67 kg respectively, and for the 'B' OTSG lower head and J-legs were 0.46 kg and 5.79 kg respectively (Reference 28). Even though the original fuel estimate for the 'A' OTSG lower head may increase significantly due to the new fluff sample data, the fuel transferred to the RV would be relatively small, significantly less than seven (7) kilograms.

The OTSGs will be drained using a TMI designed low flow (<1 gpm) pump to reduce the amount of core debris transferred during drain down. Three (3) pumps will be installed in each OTSG, one (1) in each J-leg and one (1) in the lower head of each OTSG. During a recent (February 1991) J-leg sampling program which used the actual drain pumps, it was observed that the drain pumps quickly picked up the fine debris



surrounding the intake port and then the effluent became relatively clear. It is estimated that approximately 1.0 kg of fuel ( $UO_2$ ) will be transferred from the OTSGs to the RV. This estimate assumes there is core debris within the range of the suction flow, approximately six (6) centimeters away. It also assumes that the debris is approximately one (1) centimeter deep (estimated volume =  $\pi(6)^2(1)=113 \text{ cm}^3$ ). Using an approximate debris density of  $6 \text{ g/cm}^3$ , a packing fraction of 0.5 and an average  $UO_2$  to debris ratio of 0.5 recently estimated for the J-leg fluff samples (Reference 29), the total mass of  $UO_2$  transferred from the OTSGs to the RV is estimated as 1 kg  $UO_2$  ( $6 \text{ pumps} \times 113 \text{ cm}^3 \text{ volume/pump} \times 6 \text{ g/cm}^3 \times 0.5 \times 0.5$ )

The next task is to estimate the quantity of fuel debris that will be transferred from the RV to the FTC during the sequential drain downs of the RV. While the OTSGs are being drained to the RV, it will be necessary to periodically drain the RV to the FTC to prevent the RV water level from rising too high. The water level in the RV needs to be kept well below the nozzles to prevent any chance of recirculation to the OTSGs. During the water transfer to the FTC, some (estimated 10%) of the loose fine core debris transferred from the OTSGs will be transferred to the FTC. It is assumed that 5% of the debris transferred from the OTSGs will be deposited in the pump suction area and another 5% of debris will slide into the pump suction area due to the slope of the lower head and the velocity of the water near the lower head surface. In addition to the fuel debris from the OTSGs, additional fuel debris will settle out on the lower head due to the water level sloshing in the RV during drain down of the OTSGs. It is estimated that 0.1 kg of fuel ( $UO_2$ ) would be transferred to the FTC during this sequence.

The next phase will be to completely drain the RV for the final time. While the RV is being drained it will be necessary to drain the FTC at

the same time to prevent it from overflowing. During this final drain down, it is estimated that another 1.4 kg (5% of 28 kg on lower head) could be transferred to the FTC. This would mean that an estimated total of 4.5 kg of fuel ( $UO_2$ ) will be transferred from the RV to the FTC during drain down of the OTSGs and RV. The uncertainty assigned to the total fuel debris (4.5 kg  $UO_2$ ) transferred from the RV to the FTC during drain down is  $\pm 78\%$ . This uncertainty is based on an average dimensional uncertainty of 20%, a fuel density uncertainty of 40% and a fuel debris ratio uncertainty of 53%. The value was calculated using standard propagated error formulas for a product of terms (Reference 30).

After the OTSGs and RV are completely drained, the FTC will be drained to the RCBTs using the same pumps used to drain the RV. Since the floor of the deep end of the FTC was relatively clean prior to starting drain down and the plenum covers approximately 36% of the floor area of the deep end of the FTC, only a fraction (64%) of the core debris pumped into the FTC during drain down (estimate 2.9 kg) could reach the floor of the FTC. There will be plenty of time for the fine debris to settle out in the FTC prior to draining. Based on the total FTC horizontal area (420 sq ft - Reference 31) that fuel fines could settle over and the area that the drain pump would normally suck clean (2.3 sq ft - Appendix A), it was estimated that approximately 1% of the fuel debris added to the FTC during drain down of the RV would be removed during drain down of the FTC. In order to simplify SNM Accountability, the amount of fuel removed from the FTC during drain down will continue to be included in the total quantity shown for the FTC. This will eliminate the need to revise the AFHB PDSR to account for any fuel relocated in the RCBTs during FTC drain down.

The total estimated fuel ( $UO_2$ ) remaining in the FTC after drain down is as follows:

	<u>Fuel (UO<sub>2</sub>)</u>	<u>Uncertainty</u>
Fuel washed off canisters	13.9 kg	+ 24%, -100%
Fuel on transfer tubes	0.5 kg	+ 34%, - 92%
Fuel in DWCS hoses	0.0 kg*	
Fuel added during drain down	<u>4.5 kg</u>	<u>+ 78%, - 78%</u>
Total Estimated Fuel	18.9 kg	+ 37%, - 95%

\* Fuel content in DWCS hoses was estimated at 0.012 kg UO<sub>2</sub> which rounds off to 0.0 kg.

The estimated percentage uncertainties for the measurements, calculations and estimations are shown above. Total uncertainty was determined using an asymmetrical approach where the negative and positive uncertainties were added separately and a range was determined for the final uncertainty. The range was from 2.4 kg to 26.2 kg or 18.9 kg +37%, -95%.

It is acknowledged that the method of estimating fuel transfer during drain down was relatively uncertain but since the fuel quantity transferred to the FTC was relatively small, it was decided that additional fuel measurements would not be warranted.

#### 4.3 Core Flood System

During the 1979 TMI-2 accident, fuel was deposited in the core flood system due to backflow from the reactor through one or more malfunctioning check valves. Additional fuel was deposited in the core flood tank '1A' during the defueling and removal of the lower core support assembly (LCSA). These additional fuel deposits were attributed to the following LCSA segments which were removed from the RV to permit continuous progress in the RV defueling activities.

- 13 Severed lower grid rib segments
- 4 Pie-shaped lower grid distributor segments
- 4 Pie-shaped lower grid forging segments
- 4 Incore guide support plate sections
- 12 Flow distributor plate sections
- 48 Lower grid rib support posts

Prior to removal from the RV, the LCSA segments were flushed and brushed to minimize the transfer of adherent core debris. The segments were also video inspected to ensure no visible fuel was present. Sample sections of each plate were measured by gamma spectroscopy and/or alpha measurements to determine the quantity of residual fuel. Exposure rate measurements were performed on pipe sections to establish the fuel deposited in the Core Flood System.

Exposure rate measurements were taken along the core flood system pipes using a standard HP-220A probe in 1985 and a modified HP-220A probe in 1989. Both instruments utilized a hemispherical tungsten collimator which had an approximate front to back response ratio of 50:1. The standard HP-220A had a field of view of 140 degrees, and the modified HP-220A had a field of view of 90 degrees.

Measurements were conducted by placing the probe in contact with the pipes either on the side (vertical pipes) or on the bottom (horizontal pipes). A background measurement was conducted at each measurement location by facing the probe away from the pipes. This type of background measurement estimated the general area exposure rate from other sources in the reactor building. There were two (2) other sources of background, or non-fuel related, exposure rates: Cs-137 surface contamination on the pipe surfaces and Cs-137 in the aqueous phase of the pipe contents. The average exposure rate of the

contaminated surface was 11.3 mR/hr which was used as the Cs-137 surface contamination background. The contribution to background from dissolved Cs-137 was considered to be negligible. Thus, the background subtracted from each gross exposure rate consisted of the general area background (calculated for each measurement location) and the Cs-137 surface contamination exposure (11.3 mR/hr).

Two general source geometries were used to estimate the fuel quantity in the core flood system: the fuel film in the pipes and tanks, and the fuel deposits in the pipes. The average RCS fuel film mass per unit area was multiplied by the internal surface area of the core flood system to estimate the total fuel mass in films. This portion of the estimate is conservative because the core flood system was not exposed to circulating fuel debris for as long a time as the main portion of the RCS. Thus the average fuel film concentration in the core flood system was expected to be lower than that of the RCS.

The 1985 and 1989 gamma exposure rate measurements and radionuclide concentration data from core samples were used to estimate fuel deposited in the core flood pipe segments. The observed fuel per segment distribution was then applied to the remaining, unmeasured, portion of the piping. This approach assumed that no fuel debris was deposited in the tanks or vertical pipe sections.

Because most of the "A" side piping was inaccessible, the results from the "B" side were used to estimate the fuel mass in the "A" side. The assumption is that the linear concentration of fuel in the "B" pipe is equal to that in the "A" pipe.

In addition, pieces of the lower core support assembly were removed from the reactor vessel and stored in the "A" core flood tank which had its top cut off for this purpose.

The results of the calculations for each area are summarized below and calculation details are described in Engineering Calculation 4200-3232-90-067.

	<u>A-Side</u>	<u>B-Side</u>
Fuel (UO <sub>2</sub> ) total in films	0.008 kg	0.010 kg
Fuel (UO <sub>2</sub> ) total in deposits	0.890 kg	1.670 kg
LCSA pieces	<u>2.300 kg</u>	<u>-----</u>
SUB TOTAL	3.198 kg	1.680 kg

Total Fuel Mass In Core Flood System: 4.9 kg of UO<sub>2</sub>  
 The uncertainty of this estimate is  $\pm 77\%$ .

#### 4.4 Upper Endfittings

A total of seventeen (17) upper endfittings are stored in six (6) separate storage containers. Five (5) of the endfitting storage containers are stored on elevation 347' in the Reactor Building. The sixth (6th) storage container, designated as TSC, is stored in the deep end of the fuel transfer canal. A neutron interrogation system was utilized to determine the amount of fuel remaining on the upper endfittings. The interrogation system utilized an antimony-beryllium (Sb-Be) photoneutron source and He-4 detectors. The Sb-Be source primarily produces neutrons with energy of  $24 \pm 2$  keV. The half-life of a Sb gamma source is sixty (60) days. The measurements were conducted in the Reactor Building where the presence of other equipment caused electronic noise to interfere with the counting system. The noise problem was partially overcome by performing repetitive counts and then applying Chavenets' criterion (References 30 and 32) to the resultant data.

While each endfitting storage container was placed in the counting location, the background was measured using He-4 detectors without any Sb-Be source. Then known  $UO_3$  calibration spikes were added to the storage container, and the Sb-Be source was introduced to produce interrogating neutrons. These neutrons produced fission in the U-235 in the residual fuel. Measurements were taken for all three (3) conditions. First, without the Sb-Be source; second, with the Sb-Be source; and third, with the Sb-Be and  $UO_3$  sources. The fast neutron flux from fission of U-235 was measured using the three (3) He-4 neutron detectors and associated electronic equipment. Most of the fission neutrons had energies between one (1) to two (2) MeV which was much higher than the interrogating source neutrons.

The neutron interrogation system was calibrated for each geometry encountered by placing two (2) uranium calibration spikes into the containers with the endfittings. The spikes were cylindrical containers of natural uranium with known masses. The system efficiency factors were calculated from the increase in neutron count rate per U-235 mass in the spikes for counting geometry.

The data was collected for each storage container using three (3) He-4 detectors. Counting of the signals was performed with a computer using LABTECH NOTEBOOK software. This software collected the data for each preset interval and summed the results over a preset time.

Five (5) storage containers were measured using this technique while the sixth (6th) storage container, designated as 'TSC', was analyzed using the average fuel value obtained while counting the five (5) storage containers described above. The results of the measured and estimated residual fuel quantities were as follows:

<u>Storage Container</u>	<u>Number of Endfittings</u>	<u>Fuel Quantity</u>
#7	2	0.35 kg
#9	3	0.69 kg
#5	4	3.12 kg
#4	3	0.57 kg
#6	2	0.12 kg
TSC	<u>3</u>	<u>1.04 kg</u>
Totals.	17	5.89 kg

The fractional uncertainties for the measurement and calculations were estimated as follows:

1. Calibration of Spike Masses: negligible
2. System Efficiency:  $\pm 57\%$
3. Measurement Error:  $\pm 63\%$

Total uncertainty was determined by quadrature error propagation which was the square root of the sum of the squares of both the efficiency and measurement errors.

$$\text{Sqrt } ((57\%)^2 + (63\%)^2) = \pm 85\%$$

The total fuel estimate for all of the endfittings stored in the Reactor Building was 5.9 kg of  $\text{UO}_2$ . The uncertainty associated with this estimate was  $\pm 85\%$ . The results of the detail calculations for the endfitting storage containers are described in Engineering Calculation 4249-3232-90-079.

#### 4.5 Tool Decon Facility (TDF)

The tool decon facility (TDF) consisted of three (3) rooms which were



located on the 347' level of the reactor building. The TDF had been used until January 31, 1990, for decon work; thus, only general area surveys dated after January 31, 1990, were used to determine residual fuel. The background gamma radiation level in the TDF due to general area Cs-137 contamination on the floor, walls and components was estimated as 15 mR/hr. It was assumed that negligible quantities of fuel were attached to the walls and ceilings of the TDF.

The Cs-137 to UO<sub>2</sub> ratio for core debris was taken to be 1400 ± 500 μCi Cs-137/g as of March 28, 1988 (Reference 9). Decay correction to the date of the area surveys (March 1, 1989) was neglected. Data for Co-60 and other gamma-ray emitters were not analyzed; however, the contribution of these radionuclides to the gamma exposure rate was estimated to be less than 20% of that due to Cs-137 alone.

The fuel in the TDF was assumed to be uniformly distributed over the floors and grating in the three (3) rectangular rooms. Readings in the left room (Area -1) were fairly uniform with the center reading 15 mR/hr. Thus, a model using a 12' x 12' rectangular area source with the dose point in the center was entered into Microshield (Reference 8) to calculate an exposure rate for the Area -1 fuel concentration factor mR/hr/(mg/cm<sup>2</sup>). The actual Area -1 fuel concentration (mg/cm<sup>2</sup>) was calculated by dividing the actual exposure rate (after background subtraction) by the Microshield factor. Multiplication of the Area -1 concentration by the floor area resulted in an estimate of less than 0.1 grams of fuel (UO<sub>2</sub>) on the floor.

The fuel estimate for the middle room (Area -2) was calculated using the same approach. The resulting estimate for Area -2 was 5 grams of fuel (UO<sub>2</sub>).

The right room (Area -3) had more variable exposure rates. Thus, Area

-3 concentration estimates based on each measurement point were calculated using Microshield. The four (4) corner points were assumed to be two (2) feet from one wall and three (3) feet from the other wall. The two (2) center points were assumed to be six (6) feet from the side walls and four (4) feet from the end walls. The average of the six (6) area concentration estimates multiplied by the floor area then resulted in a fuel estimate of 0.11 kg of  $UO_2$  for the room.

The total fuel estimate for the entire TDF was 0.1 kg of  $UO_2$ . The uncertainty associated with these estimates was  $\pm 100\%$ . The results of the detail calculations for each area are described in Engineering Calculation 4249-3232-90-068.

#### 4.6 Drain Line From TDF

There are numerous drain lines throughout the reactor building. Only one line was expected to have significant quantities of fuel; the line that services the defueling tool decon facility on the 347' elevation. This line begins at a floor drain in the tool decon facility, passes through the floor and extends ten (10) feet nearly horizontally to a riser. The riser drops down through the 305' level via vertical and forty-five (45) degree runs and empties into a horizontal run within the basement floor slab. The horizontal run in the basement slab extends approximately one-hundred and twenty (120) feet via a gradual slope into the RB sump. The basement had been under at least eight (8) inches of water since the accident, thus this horizontal run had been filled with water.

Other lines in the reactor building have not serviced fuel-related activities and thus are assumed to contain negligible quantities of fuel.

The fuel estimates in the drain line were made using gamma exposure rate measurements within the pipe. A Cs-137 to fuel mass ratio (Reference 9) was utilized to determine the exposure rate per unit mass of fuel within a pipe segment. The quantity of fuel within each pipe segment was then calculated using the exposure rate data and the exposure rate per unit mass fraction.

The exposure rate measurements were made using a miniature or "peanut" GM detector positioned at regular intervals within the drain pipe. Six detectors were included in the detector string, however, only detectors #1, #3 and #5 were used due to the availability of readout electronics. Detector #1 was surrounded with a cadmium energy compensation shield and was connected to a Ludlum 2200 single channel analyzer. Detectors #3 and #5 had no shield and were connected to Eberline SAM-II single channel analyzers. Detectors #3 and #5 experienced an unusually large amount of dead time during the measurements thereby nullifying the results. Only the data from detector #1 was used to calculate fuel.

The GM detectors were housed within a one-hundred (100) foot long by one-half (1/2) inch OD polyethylene tube which was water tight. The tube was attached to a plumber's snake (1/8" x 1" steel band) to cause it to lay flat in the bottom of the pipe. The detector tube insertion was incremented by one (1) and four (4) foot groups so that the detectors at one (1), three (3), and five (5) feet would provide count rate readings at each foot. However, because detectors #3 and #5 were later found to be malfunctioning, only the readings from detector #1 were used to establish the fuel mass.

The fuel estimates were obtained by subtracting the background exposure rate (5.2 R/hr) from each section and dividing by the Microshield exposure rate/mass factor (0.21 R/hr-g UO<sub>2</sub>). The total

mass identified in the eight (8)-inch drain line under the basement floor was 2.5 kg of  $UO_2$ . The total mass in the six (6)-inch drain line under the 347' floor was 1.9 kg of  $UO_2$ . This results in a total fuel mass of 4.4 kg of  $UO_2$  in the TDF drain line. The uncertainty of this total mass was  $\pm 87\%$  and includes the uncertainties due to calibration, geometry and shielding models used, and the variability in the Cs-137 to fuel ratio. See Engineering Calculation 4249-3232-90-073 for details of the data reduction.

#### 4.7 Defueling Water Cleanup System

The Defueling Water Cleanup System (DWCS) was installed during the TMI-2 Cleanup Program to remove fuel fines from the RCS water and improve visibility in the RV during defueling. The DWCS System consisted of circulating water pumps, filter canisters, a valve manifold and interconnecting hoses. All of the filter canisters have been shipped offsite for permanent storage.

The circulating water pumps, DWC-P-2A and DWC-P-2B were located in the new fuel pit in the southeast corner of the fuel transfer canal. A video inspection of the DWC-P-2A pump intake area was conducted on September 22, 1989. A depth of approximately 0.3 centimeters of fuel debris was estimated on the horizontal, annular surface of the transition area. It was assumed that DWC-P-2B contained an equal quantity of fuel. To estimate the fuel in the two (2) pump transitions, the depth (0.3 cm) was multiplied by the horizontal area ( $852 \text{ cm}^2$ ) of both transitions times the excore fuel density (4.7 g/cc) times the  $UO_2$  to debris ratio (0.72). The results of this product was the total quantity (0.86 kg  $UO_2$ ) of fuel residing on the horizontal surfaces of both pump transition areas.

Two other locations in the pumps were identified in the video inspection as containing fuel debris. The quantity was estimated as 98 and 131 cubic centimeters totaling  $(229 \text{ cm}^3 \times 4.7 \times 0.72) = 0.78 \text{ kg UO}_2$  in each pump assembly. The total quantity of fuel estimated for both DWCS pumps was  $(0.86 \text{ kg} + (2 \times 0.78 \text{ kg}))$  or  $2.4 \text{ kg of UO}_2 \pm 100\%$ .

The manifold consisted of piping, valves and other components that interfaced between the RV and filter canisters. It was located at the north end of the FTC. A series of gross gamma measurements were taken from each D-Ring using a hexadirectional probe. This probe utilized a hemispherical tungsten collimator which had an approximate front to back response of 50:1 and a field of view of 90 degrees. The background measurements consisted of pointing the detector away from the component, up and/or towards a wall of the RB. Thus the exposure rate detected was due to non-fuel related Cs-137 contamination on the RB surfaces. The resulting fuel related exposure rate for the manifold was the gross reading minus the average background readings collected on the D-Rings. The average background exposure rates for the "A" and "B" D-Rings were 5.05 and 2.37 mR/hr, respectively. The net exposure rates from the "A" and "B" D-Rings were 5.48 and 6.59 mR/hr, respectively.

The average net exposure rate approximately twenty-four (24) feet from the center of the manifold was  $((5.48 + 6.59)/2) = 6.04 \text{ mR/hr}$ . The exposure rate per unit mass factors for line and point source models with varying shielding thicknesses were calculated using the computer program Microshield (Reference 8). The results using the above data and the most probable shielding and geometry model (the best case) estimated the fuel quantity in the manifold as  $1.1 \text{ kg UO}_2 \pm 50\%$ .

The third and final component that made up the DWC System was the interconnecting hoses between the pumps, filters and manifold. Hose

sizes varied from 0.5 to 4 inches inside diameter and a total length of approximately 6550 feet. A series of gamma spectrometry and gamma exposure rate measurements were taken on one 1 foot samples and other hoses in place. The average fuel mass for the one 1 foot long hose samples was determined as  $0.015 \text{ g} \pm 90\%$  for the two (2) inch inside diameter hose which had 486 square centimeters of inside area per foot. The total inner surface area for all the DWCS hoses was determined as 614,986 square inches. Taking the product of the average fuel mass per area times the total area resulted in a fuel estimate of  $0.122 \text{ kg} \pm 90\%$  for all the hoses in the DWC System. A second method using gross gamma measurements was utilized to measure the hoses which resulted in a fuel estimate of 0.295 kg. Then the two (2) estimates were averaged to arrive at the final fuel estimate of  $0.21 \text{ kg UO}_2 \pm 100\%$ .

Details of the data analysis and measurements are contained in the Engineering Calculation No. 4249-3232-090-078.

After the DWC System had been measured, sections of the two (2)-and three (3)-inch diameter hoses were removed from the work platform and placed in the FTC. The quantity of fuel remaining on these hoses was estimated (see Section 4.2) as  $0.012 \text{ kg UO}_2 \pm 100\%$ . This amount was deducted from the total residual fuel deposits in all the hoses discussed above. The final estimate for the remaining hoses was  $(0.21 \text{ kg} - 0.01 \text{ kg})$  or  $0.20 \text{ kg UO}_2 \pm 100\%$ . The final estimate for the DWC System in the RB was:

<u>Component</u>	<u>Qty of Fuel(UO<sub>2</sub>)</u>	<u>Uncertainty</u>
DWC Pumps	2.4 kg	$\pm 100\%$
Manifold Assembly	1.1 kg	$\pm 50\%$
Interconnecting Hoses	<u>0.2 kg</u>	<u><math>\pm 100\%</math></u>
Total	3.7 kg	$\pm 67\%$

#### 4.8 Temporary RV Filter System

Curie estimates were performed on the diatomaceous earth (DE) collection drums and filters that were used in the Temporary Reactor Vessel Filtration System (TRVFS). The isotopic analysis used to determine the residual fuel came from one of the collection drums and was considered typical of the DE in the filters. The Radiological Engineering Calculation, Reference 33, was assumed to be a valid estimate of the curie loading for the two filters. The calculation was rechecked by Fuel Measurements and Analysis (FM&A) and was found to be acceptable.

Using the data provided in Reference 28, the fuel loading for both filters was estimated as 3.80 kg and 0.59 kg of  $UO_2$ . This resulted in a total of 4.4 kg of  $UO_2$  for the TRVFS. See Engineering Calculation 4800-3520-89-076 for details of the data reduction. The uncertainty associated with these values is  $\pm 90\%$  to a one-sigma confidence level.

#### 4.9 Incore Guide Tubes

The flow distributor assembly (FDA) was cut up into twenty-six (26) pieces. It was determined that thirteen (13) plates each contained one to four incore instrument guide tubes (IIGTs). The plates were removed from the reactor vessel (RV) one at a time and moved to a measurement location on top of the 'A' D-ring.

The gamma-ray spectrometry measurements were performed using a lead shielded HPGe detector. The detector assembly was connected to a preamplifier, preamplifier power supply, amplifier, and a multi-channel analyzer (MCA). The detector system was calibrated using Ce-144 and Eu-154 sources for a full energy peak calibration. The

detector system was also source checked before each measurement to ensure proper operation using a standard Ce-144 source. The quantity of reactor fuel was based upon using Ce-144 as an analog for fuel ( $UO_2$ ). The Ce-144-to-fuel ratio of  $152.5 \mu\text{Ci/g}$  ( $\pm 53\%$ ) was available on 8/1/87 (Reference 10). This activity was decayed to the date that the measurements were performed.

The calculation of the amount of fuel on each plate was performed using either the computer code QAD (Reference 34) or MICROSSHIELD (Reference 8). The source region of the IIGTs was modeled as a volume inside the IIGTs to simplify the calculation. The source was assumed to be uniformly distributed in the source region. The maximum quantity of fuel in one IIGT was (assuming a density of  $5 \text{ g/cm}^3$  ( $UO_2$ )) 1.42 kg. Two of the IIGTs, K-5 and M-10, were cut out of the flow distributor plates, stored in the 'A' D-ring and shipped offsite to a laboratory for analysis. This work was performed for the Nuclear Regulatory Commission. Both IIGTs were assayed as containing 1.42 kg of  $UO_2$  each and thus, 2.8 kg was subtracted from the 23.8 kg of fuel measured on the flow distributor plates stored in the 'A' D-ring. The adjusted fuel quantity remaining on the flow distributor plates was 21.0 kg of  $UO_2$ . The estimated uncertainty in this fuel value was  $\pm 54\%$ . Details of the data analysis and measurements are contained in Reference 35.

#### 4.10 Defueling Tools

Five (5) defueling tools were measured utilizing gamma-ray spectrometry and an HPGe shielded detector. The quantity of fuel was based upon the activity of Ce-144 and the known Ce-144-to-fuel ratio of  $152.5 \mu\text{Ci/g}$  decayed to the date the measurements were performed. The detector system was source checked before each measurement and after the final sequence of measurements to ensure proper operation



using a standard Ce-144 source. The calculation of the quantity of fuel on each tool was performed using either the computer code QAD (Reference 34) or MICROSIELD (Reference 8). Computer modeling of source geometry was used to calculate the gamma flux per unit of fuel mass on each tool. The final mass estimate for each tool was then determined by dividing the measured gamma flux by the gamma flux per unit mass factor. The total residual fuel for the five (5) measured tools was 61 grams  $\pm$  60% of UO<sub>2</sub>.

The five (5) measured tools were also measured using the gamma exposure rate method to arrive at an approximate fuel estimate for the remaining one-hundred and twelve (112) tools to be assayed. This data resulted in a gamma exposure rate-to-fuel ratio of 0.094 (R/hr)/gram. Sixty-nine (69) tools were measured using the gamma exposure rate radiation levels and were found to contain a total fuel mass of 318 grams  $\pm$  70% of UO<sub>2</sub>.

Combining this data with the data above, results in an average fuel deposit of 5.1 grams/tool of UO<sub>2</sub>. There were forty-three (43) other tools that were not measured due to radiological restrictions. The exact number of tools or tool pieces is uncertain due to multiple component tools, tools that have been cut-up, and the lack of data on some tool storage areas. Thus, without additional data, the conservative estimate of 117 total tools was used for this report. The number of tools, for which no actual or specific survey data is available, was taken as 117-74 or 43. The best estimate for the quantity of fuel on these tools is 43 x 5.1 g/tool or 220 grams  $\pm$  85% of UO<sub>2</sub>. Adding this to the 74 measured tools (379 g), results in approximately 0.6 kg  $\pm$  75% of UO<sub>2</sub> for all one-hundred and seventeen (117) defueling tools. Details of the uncertainty analysis are discussed in Engineering Calculation 4240-3232-90-066.

## 5.0 CONCLUSION

The estimate of record of the quantity of fuel ( $UO_2$ ) remaining in the Reactor Building Miscellaneous Components is 64.0 kilograms, (at one sigma). This uncertainty is based on the individual component uncertainties associated with each component addressed in this Post Defueling Survey Report. The range of the estimate of record extends from 31.6 to 85.4 kg  $UO_2$ .

This estimate of record is derived from the measured and analyzed data summarized in Table 2. The data shows that 62% of the remaining fuel is contained in two components, the Fuel Transfer Canal/Transfer Tubes and the Incore Guide Tubes stored in the 'A' D-Ring. These components are expected to remain static once drain down is completed. No future water flows are planned except possibly to remove condensation from the FTC. Additional measurements of the RB Miscellaneous Components are not considered to be justifiable based on ALARA considerations due to the small quantity of fuel ( $UO_2$ ) measured to date. After final draindown of the RCS, the RB will be normally isolated from all systems outside the RB, and any operation that could result in a potential fuel movement will be closely controlled.

The goal of the TMI-2 defueling program was to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the 64.0 kg quantity of  $UO_2$  remaining in the RB Miscellaneous components is less than 10% of the total fuel ( $UO_2$ ) remaining at TMI-2.

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TABLE 1

ASSAY METHODS UTILIZED TO MEASURE  
MISCELLANEOUS AREAS IN THE REACTOR BUILDING

AREA DESCRIPTION	ASSAY METHOD	CALCULATION NUMBER
Reactor Coolant Drain Tank	Sampling & Analysis	Reference 13
Fuel Transfer Canial/Transfer Tubes	Gamma Dose Rate	4800-4420-89-095
Core Flood System	Gamma Dose Rate	4200-3232-90-067
Upper Endfittings	Neutron Interrogation	4249-3232-90-079
Tool Decon Facility (TDF)	Gamma Dose Rate	4249-3232-90-068
Drain Line From TDF	Gamma Dose Rate	4249-3232-90-073
Defueling Water Cleanup System	Gamma Spec./Gamma Dose Rate	4249-3232-90-078
Temporary RV Filter System	Sampling & Analysis	4800-3520-89-076
Incore Guide Tubes	Gamma Spectroscopy	4200-3211-90-062
Defueling Tools	Gamma Spec./Gamma Dose Rate	4240-3232-90-066

TABLE 2  
SUMMARY OF SNM INVENTORY FOR RB MISCELLANEOUS AREAS

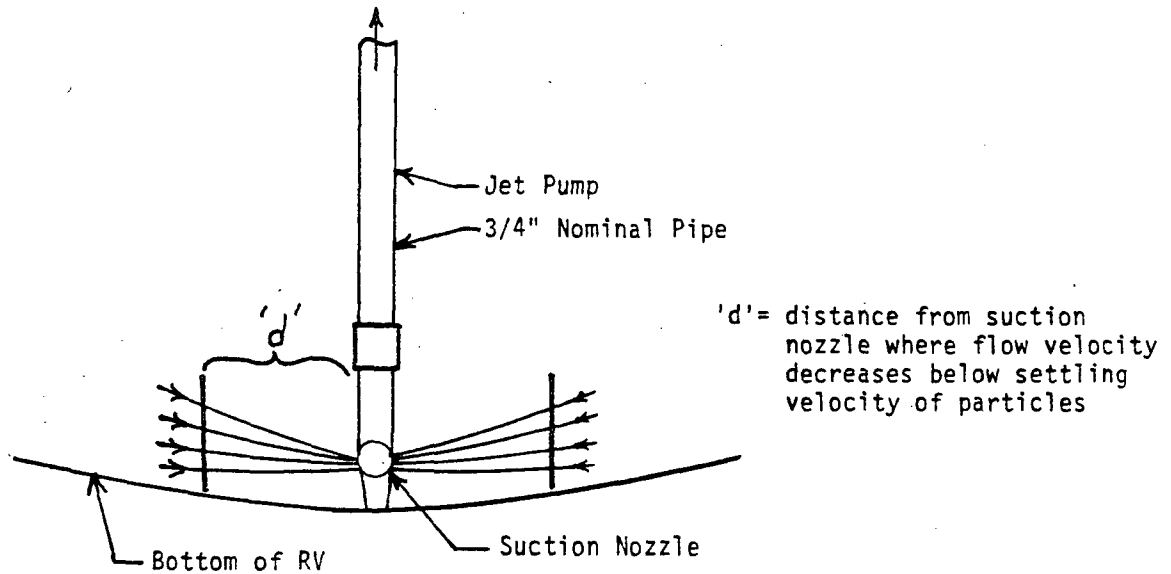
<u>AREA DESCRIPTION</u>	<u>FUEL QUANTITY</u>	<u>UNCERTAINTY</u>	<u>REFERENCE</u>
<u>Symmetrical Data:</u>			
Reactor Coolant Drain Tank	0.1 kg	± 54%	PDSR Paragraph 4.1
Core Flood System	4.9 kg	± 77%	PDSR Paragraph 4.3
Upper Endfittings	5.9 kg	± 85%	PDSR Paragraph 4.4
Tool Decon Facility (TDF)	0.1 kg	±100%	PDSR Paragraph 4.5
Drain Line From TDF	4.4 kg	± 87%	PDSR Paragraph 4.6
Defueling Water Cleanup System	3.7 kg	± 67%	PDSR Paragraph 4.7
Temporary RV Filter System	4.4 kg	± 90%	PDSR Paragraph 4.8
Incore Guide Tubes	21.0 kg	± 54%	PDSR Paragraph 4.9
Defueling Tools	<u>0.6 kg</u>	<u>± 75%</u>	PDSR Paragraph 4.10
Sub-Total	45.1 kg	± 32%	
<u>Asymmetrical Data:</u>			
Fuel Transfer Canal/Transfer Tubes	<u>18.9 kg</u>	± 37%, -95%	PDSR Paragraph 4.2
ESTIMATE OF RECORD (UO <sub>2</sub> )	64.0 kg		
TOTAL UO <sub>2</sub> RANGE: Low Value: 31.6 kg, High Value: 85.4 kg			

NOTE: The range was determined using the asymmetrical term for the Fuel Transfer Canal/Transfer Tubes and combining its range algebraically with the fuel quantity (45.1 kg UO<sub>2</sub> ± 32%) estimated for the symmetrical terms.



## APPENDIX A

### Pumping Velocity at Known Distance from Pump Suction Nozzle



Assumed Pumping Rate = 10 GPM

Water Velocity in 3/4 " pipe = 6.02 ft/sec or 72.2 in/sec.

Cross section area of 3/4 " pipe = 0.533 sq. in.

Assumed surface area of cylinder at distance 'd' from jet pump which surrounds pump suction nozzle for a uniform flow velocity =  $\pi d^2$

Effective settling Area for lower vessel head = 112 sq. ft.

Assumed water velocity ' $v_s$ ' at distance 'd' from suction nozzle:

$$v_s = \left( \frac{\text{area of pipe}}{\text{area of surface}} \right) (\text{velocity in pipe})$$

$$v_s = \frac{0.533 \text{ in}^2}{\pi d^2 \text{ in}^2} (72.2 \text{ in/sec}) = \frac{12.25}{d^2} \text{ in/sec}$$

(1)

at  $d = 6 \text{ cm}$ ;  $V_s = 2.2 \text{ in/sec}$  or  $5.6 \text{ cm/sec}$

5.6 cm/sec is the approximate settling velocity for particle sizes less than 500 microns.

(2)

at  $d = 25.4 \text{ cm}$  or  $10 \text{ inches}$ ;  $V_s = 0.12 \text{ in/sec}$  or  $0.30 \text{ cm/sec}$

0.30 cm/sec is the approximate settling velocity for particle sizes less than 70 microns

Area enclosed by 'd' radius = 2.3 sq. ft.

Approximate percentage of lower head settling area = 3%

\* \* \* \* \*

Assumed fuel debris particle size distribution based on Lower Vessel SSC-1-2 Sample data with no particles larger than one millimeter (Reference 36).

<u>Particle Size</u>	<u>Cummulative Percent Weight</u>	<u>Maximum Settling Velocity*</u>
< 38 micron	10%	0.07 cm/sec
< 90 micron	30%	0.5 cm/sec
< 150 micron	46%	1.3 cm/sec
< 300 micron	67%	3.0 cm/sec
< 710 micron	91%	7.0 cm/sec
<1000 micron	100%	12.0 cm/sec

\* Settling velocities determined from CRC Handbook of Chemistry and Physics, 68th Edition (Page F-233).

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GPU Nuclear Corporation  
Post Office Box 480  
Route 441 South  
Middletown, Pennsylvania 17057-0191  
717 944-7621  
TELEX 84-2386  
Writer's Direct Dial Number:

(717) 948-8400

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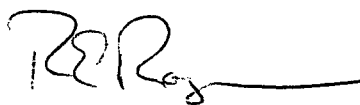
Dear Sir:

By NRC letter dated October 17, 1985, GPU Nuclear was granted exemption from certain requirements for periodic inventory and reporting of the special nuclear materials (SNM) balance for Three Mile Island Unit 2 (TMI-2). As a condition of the exemption, GPU Nuclear is required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment is referred to in the exemption as the "post-defueling survey." GPU Nuclear letter 4410-88-L-0162 dated September 30, 1988, submitted the initial Post-Defueling Survey Reports (PDSRs).

As stated in that submittal, the PDSR documents the GPU Nuclear assessment of the amount of residual SNM in the various facilities, systems, and components of the plant and describes the methodology utilized to determine the quantity of SNM in each case. The attached PDSR transmits the post-defueling survey results for the Auxiliary and Fuel Handling Buildings.

The remaining PDSRs will be submitted as they are completed. A compilation of the individual PDSRs will form the basis for a final assessment of the quantity of residual SNM at TMI-2 for accountability purposes.

Sincerely,

  
for R. L. Long  
Director, Corporate Services/TMI-2

EDS/dlb

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*APOL*

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Page 2


cc: T. T. Martin - Regional Administrator, Region I  
M. T. Masnik - Project Manager, PDNP Directorate  
L. H. Thonus - Project Manager, TMI  
F. I. Young - Senior Resident Inspector, TMI

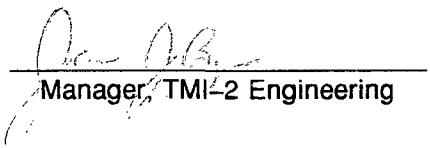


**TMI-2 POST-DEFUELING SURVEY REPORT**

**FOR THE**

**AUXILIARY & FUEL HANDLING BUILDINGS**

APPROVED:   
Manager, TMI-2 Licensing

APPROVED:   
Manager, TMI-2 Engineering

TMI-2 INTERNAL REVIEWS

AUTHOR:

F. W. Deininger 5/31/91  
F. W. Deininger

RESPONSIBLE TECHNICAL REVIEWERS:

Adam W. Miller 5/31/91  
A. W. Miller

J. S. Schork 5/31/91  
J. S. Schork

MANAGER, FUEL MEASUREMENTS & ANALYSIS:

Carl H. Distenfeld 6/3/91  
C. H. Distenfeld

PDSR COMMITTEE CHAIRMAN:

R. E. Rogan  
R. E. Rogan

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### SUMMARY

The estimate of record of the amount of uranium dioxide (UO<sub>2</sub>) remaining in the Auxiliary and Fuel Handling Buildings (AFHB) is 11.5 kg with a range from 4.2 to 15.8 Kg, distributed approximately as follows:

Spent Fuel Pool - A (FH109) . . . . .	3.8 kg UO <sub>2</sub>
RC Bleed Holdup Tanks - B&C (AX020) . . .	3.5 kg UO <sub>2</sub>
MU&P Demineralizer - 1A (AX114) . . . . .	1.1 kg UO <sub>2</sub>
MU&P Demineralizer - 1B (AX115) . . . . .	0.1 kg UO <sub>2</sub>
MU Suction Vavles (FH001) . . . . .	0.5 kg UO <sub>2</sub>
Seal Return Coolers (AX112) . . . . .	0.3 kg UO <sub>2</sub>
Makeup Tank (AX116) . . . . .	0.3 kg UO <sub>2</sub>
RC Bleed Holdup Tank - A (AX021) . . . . .	0.3 kg UO <sub>2</sub>
MU&P Valves (FH101) . . . . .	0.3 kg UO <sub>2</sub>
Makeup Pump - 1A (AX007) . . . . .	0.2 kg UO <sub>2</sub>
78 other AFHB Areas . . . . .	<u>1.1 kg UO<sub>2</sub></u>
AFHB Total	11.5 kg UO <sub>2</sub>

The above fuel distribution table shows that 63% of the UO<sub>2</sub> remaining in the AFHB is located in two (2) areas, the Spent Fuel Pool A (FH109) and the RC Bleed Holdup Tanks B&C (AX020). The majority of this 7.3 kg of UO<sub>2</sub> is in the form of silt/sediment which was deposited in its present location after flushing or washing fuel laden pipes or canisters. In determining the quantity of fuel present in each area, a system wide approach was taken. Based on this review, it was determined that approximately 67% of the residual fuel was located in two (2) systems, Makeup & Purification and Liquid Radwaste Disposal Systems, and the majority of the remainder of the 11.5 kg was contained in the Spent Fuel Pool A.

The AFHB is divided into one-hundred and thirty-one (131) individual areas covering five (5) elevations. Four (4) areas are

located on elevation 258'6" and forty-two (42) areas are located on elevation 280'6", see Figure 1. Forty-six (46) areas are located on elevation 305'0", see Figure 2. Twenty-eight (28) areas are located on elevation 328'0", see Figure 3. Eleven (11) areas are located on elevation 347'6" or above, see Figure 4.

Thirty-eight (38) individual areas in the AFHB were measured. Twenty-eight (28) areas were measured using Gamma Scintillation, four (4) areas were measured using sample analysis and the remaining six (6) areas using Gross Gamma Exposure Rate measurements. Another fifty (50) areas were analyzed for fuel ( $UO_2$ ) based on their vulnerability during the accident and their service during the cleanup program at TMI-2. Forty-three (43) other areas were evaluated and were judged as not containing any fuel because they were not connected in any way to the process lines supporting the reactor coolant system, the spent resin storage system or the RB sump pumping system.

The estimate of record, 11.5 kg  $UO_2$ , represents approximately 1½% of the total fuel remaining at TMI-2. The AFHB is expected to remain static since most of the systems/areas have been drained and isolated, and the remaining active systems will not be subjected to the large flow rates necessary to transport fuel. Additional measurements in the AFHB are not considered to be justifiable based on the measurement results to date and the potentially very small quantity of fuel ( $UO_2$ ) involved. After final draindown of the RCS, the AFHB will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

TMI-2 POST-DEFUELING SURVEY REPORT  
FOR  
THE AUXILIARY AND FUEL HANDLING BUILDINGS

1.0 INTRODUCTION

This report presents the analysis of the amount of  $UO_2$  remaining in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are shown in the floor plan elevation drawings included with this report as Figures 1, 2, 3 and 4. This report is one in a series of reports prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 1). All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the AFHB and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the measurement methods used to assay the residual fuel ( $UO_2$ ) in each area in the AFHB containing process systems which support and/or interface with the primary reactor coolant system. The majority of areas were assayed using gamma spectroscopy featuring a shielded thallium activated sodium iodide (NaI) scintillation detector connected to a multichannel analyzer (MCA). Since the presence of Cerium-144 (Pr-144) was not always identifiable, a minimum detectable level (MDL) calculation was prepared when appropriate.

Section 4, "Analysis", explains the methodology for arriving at the estimate of record of fuel ( $UO_2$ ) in the AFHB using multiple measurements and discusses supporting data, assumptions made and calculations used.

Section 5, "Conclusion", presents the estimate of record, uncertainty and MDL values for the amount of SNM remaining in the AFHB and is based upon the available data.

## 2.0 BACKGROUND

The March 1979 TMI-2 accident resulted in significant damage to the reactor core and subsequent release of fuel particles and fission products into the Reactor Coolant System (RCS) and other closely interconnected systems. The damaged core consisted of loose fuel pellets, solidified fuel, structural metal components, loose rubble, and partial fuel assemblies. Therefore, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

The Auxiliary and Fuel Handling Buildings (AFHB) are divided into two parts that are separated by a common wall. One part, the Auxiliary Building, contains tanks, pumps, piping and other equipment for the processing and storage of water for the reactor and primary cooling system and for the treatment of radioactive wastes. The other part, the Fuel Handling Building, contains fuel handling and storage equipment and a limited number of tanks, filters, pumps and coolers, and other facilities. The general layout of the AFHB is shown in Figures 1, 2, 3 and 4. The interior of the AFHB was contaminated by radioactive material as a consequence of the accident and cleanup activities. Piping systems that interfaced with the RCS were also contaminated with radioactive water and/or core debris. Early estimates (Reference 2) reported that approximately 25 kg of  $UO_2$  was transported to the AFHB during the accident sequence. In addition, certain defueling water processing activities were considered to have transported as much as an additional 15 kg of fuel from the RB into the AFHB for a total of 40 kg. Subsequent decontamination and water processing have reduced the AFHB residual fuel inventory to its current level of approximately 12 kg ( $UO_2$ ).

The majority of the twenty-six (26) systems in the AFHB were either never contaminated or have been flushed and drained in preparation for



post-defueling monitored storage (PDMS). Most of the remaining fuel ( $UO_2$ ) is contained within two systems, the Makeup and Purification System and the Liquid Radwaste Disposal System, and is located in various components, tanks and pipes. This residual fuel is believed to be in the form of finely divided particles and sediment material with some minor amounts of fuel remaining as adherent films. The quantity of fuel ( $UO_2$ ) is substantially less than the TMI-2 safe fuel mass limit (SFML) of 140 kg for fuel external to the Reactor Vessel (Reference 3); thus, within the AFHB, there is no potential for fuel accumulation which could result in a critical mass.

After draindown of the RCS and spent fuel pools, including the reactor vessel, the AFHB will be isolated from the Reactor Building and other miscellaneous buildings by administrative controls to ensure that containment isolation is maintained and primary system valves remain in the closed position except when authorized to be open to process individual batches of water on an as needed basis.

### 3.0 MEASUREMENT METHODS

Post-defueling SNM measurement at TMI-2 was a complex task. Several different measurement techniques were used. Technique selection for a particular area depended on the configuration of the area, structure or component to be assayed, physical access limitations, area radiation dose rates and the likely nature of the form and quantity of fuel deposits. In the AFHB, the measurement techniques were limited to Gamma Scintillation Counting for all except ten (10) areas. In the ten (10) areas where Gamma Scintillation was not used to measure fuel quantities, gross gamma exposure rates were measured in six (6) areas. In four (4) other areas fuel content was established using individual samples (fuel/resin) from each area.

Extrapolations were made of residual fuel in twenty-nine (29) areas that were based on measurements taken in other areas subjected to similar accident conditions. Generally the residual fuel remaining in those areas not specifically measured was assessed to be small, i.e., less than 300 grams, and insignificant in terms of the total fuel remaining at TMI-2. The results of every area measured are supported by a formal Engineering Calculation referenced in Table 1. Each Engineering Calculation generally contains a preapproved measurement plan including both test equipment setup and data that were collected during the measurement period, analysis of the data/measurement recorded and the results of the individual assay. In December 1986, the implementing procedure was formalized to assure that the measurements were performed in a proper, accurate, verifiable and approved manner and is described in References 4 and 5.

The following assay methods were utilized to measure thirty-eight (38) areas in the Auxiliary and Fuel Handling Buildings (AFHB).

### 3.1 Sodium Iodide (NaI) - Gamma Scintillation

The following twenty-two (22) areas were surveyed using a NaI (TI) detector and gamma spectroscopy:

<u>Area</u>	<u>Description</u>
AX006	Makeup Pump (1B) Room
AX007	Makeup Pump (1A) Room
AX011	AB Sump Pump & Valve Room
AX012	AB Sump & Tank Room
AX015a/b	Cleanup Filter Rooms
AX018	Waste Transfer Pump Room
AX019	WDL Valve Room
AX026	Seal Injection Filter Room
AX112	Seal Return Cooler Room
AX116	Makeup Tank Room
AX117	MU&P Filter Room
AX501	RB Spray Pump - 1A
AX502	RB Spray Pump - 1B
AX503	DHR Cooler & Pump - 1A
AX504	DHR Cooler & Pump - 1B
FH001	MU Suction Valve Room
FH003a/b	MU Discharge Valve Rooms
FH008	Neutralizer Tank Pump
FH009	Neutralizer Tank
FH101	MU&P Valve Room

Sodium iodide (NaI) detectors were used extensively in the AFHB measurement program. These detectors are thallium activated scintillation crystals; i.e. photon interactions in the NaI (TI) crystals produced light pulses that were detected, changed to an

electronic signal, and amplified by a photo multiplier tube. This device measured fuel by detection of the 2.19 MeV gamma ray of Cerium-144 (Ce-144). Ce-144(Pr-144) was selected as a tracer for reactor fuel and a Ce-144-to-fuel (UO<sub>2</sub>) ratio was developed using sample data (Reference 6). The 2.19 MeV gamma was the highest energy gamma-ray peak that was observable thereby allowing the use of the low resolution NaI (Tl) System. This device was initially swamped by the gamma flux. The problem was solved by using the smallest detector that would successfully collect the 2.19 MeV gamma ray and the maximum practical amount of shielding. New York University optimized the design for the TMI-2 application. It consisted of a 1/2-inch diameter by 3/4-inch long crystal, a 1/2-inch diameter photomultiplier tube and 35 to 80 pounds of tungsten shielding that could be assembled in various configurations to enhance various source/detector counting geometries. The detector had good sensitivity. However, Ce-144 decay (284 d half-life) introduced a significant loss of sensitivity over the time frame of the Recovery Program requiring a shift to a High Purity Germanium Detector System. Additional details of these measurement techniques are described in References 7 and 8.

### 3.2 High-Purity Germanium (HPGe) Spectrometer

The following six (6) areas were surveyed using high-purity germanium detector systems:

<u>Area</u>	<u>Description</u>
AX004	Seal Injection Valve Room
AX014	RC Evaporator
AX020	RC Bleed Holdup Tanks - B & C
AX021	RC Bleed Holdup Tank - A

AX218 Concentrated Waste Tank  
FH112 Annulus - Elevation 305'

The HPGe detector was used to locate and quantify the residual fuel ( $UO_2$ ) in the above mentioned areas in the AFHB. AX004 and FH112 contain portions of the Makeup and Purification System while the other four (4) areas contain portions of the Radwaste Disposal System. The HPGe detector was housed in a 2-inch thick cylindrical lead shield. The detector was connected to a preamplifier, amplifier and multi-channel analyzer (MCA) as described in Reference 9. The MCA and input-output device were at a remote location; as much as 100 feet of signal cable was connected between the amplifier and analyzer. The HPGe detector measurements relied on Ce-144 and Europium-154 (Eu-154) as analogs for the residual fuel. The ratios Ce-144-and Eu-154-to-fuel ( $UO_2$ ) were 152.5 and 42.6  $\mu Ci g^{-1}$ , respectively, as of 8-1-87 (Reference 6).

The detector systems were source checked periodically to ensure proper operation using standard Ce-144 and Eu-154 sources. After the calibration data had been collected, the standard sources were removed and the acquisition of data was performed.

### 3.3 Sample Analysis Determination

Four (4) areas (MU&P Demineralizer A (AX114), MU&P Demineralizer B (AX115), Deborating Demineralizer B (AX129), and Deborating Demineralizer A (AX130)) were evaluated based on individual samples of resin media analyzed by radiochemical laboratory. The samples taken from the MU&P Demineralizers were sent to Oak Ridge National Laboratory for evaluation and chemical analysis (Reference 10). The samples taken from the Deborating Demineralizers were analyzed on

site, and the results are shown in Reference 11.

### 3.4 GROSS GAMMA

Gross gamma directional surveys were performed on piping and components when exposure rates and accessibility permitted. Six (6) areas (Makeup Pump - 1C (AX005), AB Sump Filters (AX024), Concentrated Liquid Waste Pump (AX124), Model Room (FH105), Monitor Tanks (FH106), and Spent Fuel Pool A (FH109)) were analyzed using gross gamma directional surveys. The measurement technique utilized a portable gamma survey instrument consisting of a directional probe and readout device. TMI-2 core debris contained fission and activation products which produced gross gamma-ray exposure rate values based on the radioisotopic distribution of sample data. Using the reactor core debris sample data, a correlation was developed between the gross gamma-ray output and the quantity of fuel present (Reference 12). Computer modeling was used to predict exposure rates for a given volume of fuel under the appropriate geometry for a particular location. Field measurements were made for specific locations and compared with the computer modeled output. Comparison was accomplished on a point-by-point basis. The results were generally good and the data was used to determine the quantity of fuel present in any area or component.

#### 4.0 ANALYSIS

The quantity of residual fuel ( $UO_2$ ) in the Auxiliary and Fuel Handling Buildings (AFHB) was determined by performing a series of non-destructive measurements, as described in Section 3.0, Measurement Methods. The entire AFHB was reviewed to identify those areas that could contain fuel deposits. The Rogovin Report (Reference 13) was used as the authoritative basis for the review. For the purpose of this Post-Defueling Survey Report (PDSR), the term Special Nuclear Material (SNM) was utilized to describe the residual material derived from the original  $UO_2$  fuel and should not be construed as the amount of contained U-235. The final estimate of record of residual SNM was performed employing gamma detection measurements and engineering analyses. Engineering analyses were performed in areas where the total quantity was believed to be insignificant and where accident flowpaths were established.

An insignificant quantity of residual fuel was defined as about 7 kg of  $UO_2$  or 5% of the maximum safe fuel mass for the AFHB (Reference 3). All assessment measurements were performed using direct gamma readings or counting techniques and are supported by formal engineering calculations containing relevant data and data reduction. The result of the assessment was an "estimate of record" of residual fuel ( $UO_2$ ) within a determined range for the entire AFHB. The method used to construct the overall uncertainty is discussed in Appendix F.

Two (2) major plant systems contained 67% of the residual fuel in the AFHB. The two (2) systems are the Makeup and Purification (MU&P) and the Liquid Radwaste Disposal System (WDL). Other supporting systems contained quantities of fuel as listed in Table 7. The analysis for each plant system is discussed in the Appendices attached to this PDSR.

The eight (8) plant systems (identified in Tables 4, 5, 6 and 7) evaluated for fuel content occupy sixty-nine (69) individual areas in the AFHB and are listed in the Appendices for each system. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan, Reference 1.

Another forty-three (43) areas, shown in Table 3, were judged as not requiring SNM assessment based upon authoritative analyses of the TMI-2 accident, Reference 13. These areas contain non-water processing equipment such as electrical switchgear, unit sub-stations, HVAC blowers and ducting, or no equipment at all, and served primarily as an access corridor to service other areas. There was no means to transport fuel into these areas, e.g., no piping connected to any of the processing systems that support operation of reactor coolant system.

The final nineteen (19) areas contain no residual fuel and are discussed in detail in Appendix E. The nineteen (19) areas were assessed to be free of fuel because they were completely isolated from all fuel or had been contaminated and have since been decontaminated by repeated cleaning/flushing.

The MU&P System occupies portions of eighteen (18) areas in the AFHB which are illustrated in Figures 5 and 6. Fifteen (15) of the areas were measured for residual fuel and three (3) unmeasured areas were analyzed for fuel based on their operational history during the TMI-2 accident and their service during the recovery phase. The three (3) unmeasured areas (FH002, FH004, and FH014) were assigned a residual fuel content of 0.160 kg of  $UO_2$  which is probably a high estimate of their actual content. The entire MU&P System was determined to contain  $2.81 \text{ Kg } UO_2 \pm 27\%$  plus 0.60 Kg  $UO_2$  (MDL value). This results in a range of values from 2.05 Kg to 4.17 Kg  $UO_2$ . Details of this residual fuel appraisal are discussed in



Appendix A of this PDSR.

The WDL System occupies portions of twenty-nine (29) areas in the AFHB which are illustrated in Figures 7, 8, 9, 10 and 11 except for AX101 which is not connected to any of the processing pathways. Twenty (20) of the areas were measured for residual fuel and nine (9) unmeasured areas were analyzed for fuel based on their exposure during the TMI-2 accident and their service during the cleanup program. The nine (9) unmeasured areas (AX013, AX016, AX017, AX101, AX102, AX128, AX131, AX134 and FH012) were determined to contain 0.107 kg of  $UO_2$  which is considered a conservative value. The entire WDL System was determined to contain 4.13 Kg  $UO_2 \pm 71\%$  plus 0.11 Kg  $UO_2$  (MDL value). This results in a range of values from 1.20 Kg to 7.17 Kg  $UO_2$ . Details of this residual SNM appraisal are discussed in Appendix B of this PDSR.

The WDS System occupies portions of seven (7) areas in the AFHB which are illustrated in Figures 12 and 13. Two of the areas, AX124 and AX218, were measured for residual fuel and five (5) unmeasured areas were analyzed for fuel based on their exposure during the TMI-2 accident and their service during the recovery phase. The five (5) unmeasured areas (AX008, AX009, AX010, FH010 and FH011) were estimated to contain 0.004 kg of  $UO_2$  which is considered a conservative estimate of record of their SNM content. The entire WDS System was determined to contain approximately 0.01 kg which has been rounded to 0.0 kg of  $UO_2$  for purposes of this report. This sum is less than 0.1% of the total  $UO_2$  remaining at TMI-2. Details of this residual SNM appraisal are discussed in Appendix C of this PDSR.

Appendix D addresses a group of seventeen (17) areas where fuel deposits are probably located. One area, the Spent Fuel Pool A (FH109), underwent a survey in the flooded condition and was appraised to contain 3.8 kg of  $UO_2$ . This represents the largest single deposit in the AFHB. The balance

of the areas evaluated in Appendix D contained an insignificant quantity of  $UO_2$ . Details of the residual SNM appraised for each area are discussed in Appendix D of this PDSR.

Appendix E addresses a group of nineteen (19) areas where fuel deposits could have been located if the function of each area had been misused. No areas were formally measured and Appendix E presents the rationale supporting a determination that each area does not contain fuel ( $UO_2$ ).

This survey resulted in a total of eighty-eight (88) areas being measured or analyzed for fuel content in the AFHB. The individual appraisal for each area is shown in Tables 4, 5, 6 and 7 and Appendix E of this PDSR. Thirty-eight (38) areas were formally measured using techniques discussed in Section 3.0, Measurement Methods, while the remainder were appraised based on their involvement during the accident and service during the TMI-2 Cleanup Program. Forty-three (43) areas, listed in Table 3, were evaluated and judged not to contain fuel because they were not connected in any way to the process lines supporting the reactor coolant system.

Measurement of those areas in the AFHB that were not measured during this survey would result in additional unwarranted occupational radiation exposure to measurement teams and RadCon technicians. Review of the current analytical results indicates that measurements would not be expected to change the estimate of record by more than one kilogram and would not improve the accuracy or precision of the existing estimate of record. Additionally, the relative amount of residual fuel ( $UO_2$ ) in the AFHB, when viewed in the context of the projected total inventory of  $UO_2$ , does not warrant additional radiation exposures in the AFHB to attempt to improve the present residual fuel values.

Approximately 10% of the process piping was assumed to be embedded in

concrete walls and floors and was not included in the areas of the AFHB for which fuel was inferred. Each summary table for each system contains a contribution for embedded piping; the total contribution is 0.21 kg of  $UO_2$ .

The term "uncertainty" in this Post-Defueling Survey Report is used to represent the error of each measured "estimate of record" and is taken to be one sigma. The uncertainty was determined by the project engineer for each individual measurement and associated engineering calculation. The overall uncertainty value was derived by taking into consideration errors in mathematical modeling, measurement equipment, counting, standard values (fissionable material), randomness, and uncertainty in background levels. A range of fuel ( $UO_2$ ) values was established using the measured and estimated data, uncertainty values and MDL values in accordance with the method outlined in Appendix F.

This approach resulted in an overall AFHB estimate of record of 11.5 kg  $UO_2$  with an overall range of values from 4.22 Kg to 15.78 Kg  $UO_2$ . This estimate of record was calculated by summing the operating system totals shown in Table 2 and discussed in detail in Appendix F. All residual fuel values determined to be less than 0.005 Kg of  $UO_2$  have been rounded off to 0.00 Kg for purposes of this report. Tables 4, 5, 6 and 7 show the rounded-off values.

## 5.0 Conclusion

The estimate of record of the quantity of fuel ( $UO_2$ ) remaining in the AFHB is 11.5 kilograms. Based on the data shown in Table 2 and the data reduction methods outlined in Appendix F, the range of the estimate of record extends from 4.22 Kg to 15.78 Kg  $UO_2$ . The MDL portion of the upper range value is 0.71 Kg  $UO_2$ .

This estimate of record is derived from the measured and analyzed data summarized in Table 2. The data shows that most of the remaining fuel is contained in two systems (the Makeup and Purification System and the Liquid Radwaste Disposal System) and the Spent Fuel Pool A. The amount and location of the residual fuel in the AFHB is expected to remain static because many of the systems will be drained and isolated, and the water movement through the remaining systems will be limited. Consequently, all of the fuel that might move due to significant flow rates has already been relocated to a tank, filter or other low velocity location. Additional measurements in the AFHB are not considered to be warranted based on ALARA considerations due to the small quantity of fuel ( $UO_2$ ) estimated to be located residual to the AFHB. After final draindown of the RCS, the AFHB will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

The goal of the TMI-2 defueling program, to remove more than 99% of the original core inventory of approximately 94,000 kg, was achieved. In that context, the 11.5 Kg quantity of  $UO_2$  remaining in the AFHB is approximately 1½% of the total residual  $UO_2$  inventory for TMI-2.

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TABLE 1

**ASSAY METHODS UTILIZED TO MEASURE  
AREAS IN THE AFHB**

<u>AREA</u>	<u>DESCRIPTION</u>	<u>ASSAY METHOD</u>	<u>CALCULATION NUMBER</u>
AX004	Seal Injection Valve Room	HPGe w/Ce-144/Eu-154	4800-3211-89-103
AX005	Makeup Pump - 1C	Gross Gamma E530	4249-3232-91-001
AX006	Makeup Pump - 1B	Nal Gam Spec - Ce-144	4550-3211-88-009
AX007	Makeup Pump - 1A	Nal Gam Spec - Ce-144	4550-3211-87-027
AX011	AB Sump Tank Pumps & Valves	Nal Gam Spec - Ce-144	4200-3232-90-048
AX012	AB Sump, Pumps & Tank	Nal Gam Spec - Ce-144	4200-3232-90-048
AX014	RC Evaporator (WDL-Z-1)	HPGe	Reference 17
AX015a	Cleanup Filters WDL-F-6B/9B	Nal Gam Spec - Ce-144	4249-3232-90-070
AX015b	Cleanup Filters WDL-F-6A/9A	Nal Gam Spec - Ce-144	4249-3232-90-070
AX018	Waste Transfer Pumps	Nal Gam Spec - Ce-144	4249-3232-90-071
AX019	WDL Valve Room	Nal Gam Spec - Ce-144	4200-3232-90-049
AX020	RC Bleed Holdup Tanks - B & C	HPGe w/Eu-154	4800-3232-89-099
AX021	RC Bleed Holdup Tank - A	HPGe w/Eu-154	4800-3232-89-077
AX024	AB Sump Filters	Gross Gamma E530	4200-3232-90-064
AX026	Seal Injection Filters	Nal Gam Spec - Ce-144	4550-3211-87-017
AX112	Seal Return Coolers	Nal Gam Spec - Ce-144	4550-3211-87-025
AX114	MU&P Demineralizer - 1A	Sampled	4249-3211-90-069
AX115	MU&P Demineralizer - 1B	Sampled	4249-3211-90-069
AX116	Makeup Tank (MU-T-1)	Nal Gam Spec - Ce-144	4550-3211-87-038
AX117	MU&P Filters (F-2A/B & 5A/B)	Nal Gam Spec - Ce-144	4200-3233-90-053
AX124	Conc. Liquid Waste Pump	Gross Gamma HP220	4200-3233-90-050
AX129	Deborating Demineralizer - 1B	Sampled & used Ce-144	4550-3232-87-016
AX130	Deborating Demineralizer - 1A	Sampled & used Ce-144	4550-3232-87-016
AX218	Conc. Waste Tank (WDS-T-2)	HPGe w/Ce-144/Eu-154	4240-3233-90-061
AX501	RB Spray Pump - 1A	Nal Gam Spec - Ce-144	4249-3232-90-072
AX502	RB Spray Pump - 1B	Nal Gam Spec - Ce-144	4249-3232-90-072
AX503	DHR Cooler & Pump - 1A	Nal Gam Spec - Ce-144	4249-3232-90-072
AX504	DHR Cooler & Pump - 1B	Nal Gam Spec - Ce-144	4249-3232-90-072
FH001	MU Suction Valves	Nal Gam Spec - Ce-144	4200-3233-90-056
FH003a	MU Discharge Valves	Nal Gam Spec - Ce-144	4200-3233-90-057
FH003b	MU Discharge Valves	Nal Gam Spec - Ce-144	4200-3233-90-057
FH008	Neutralizer Tank Pump	Nal Gam Spec - Ce-144	4200-3233-90-054
FH009	Neutralizer Tanks (T-8A/8B)	Nal Gam Spec - Ce-144	4200-3233-90-054
FH101	MU&P Valves	Nal Gam Spec - Ce-144	4200-3233-90-055
FH105	Model Room	Gross Gamma HP220A	4200-3551-90-051
FH106	Monitor Tanks (SDS-T-1A/1B)	Gross Gamma HP220A	4200-3551-90-051
FH109	Spent Fuel Pool A	Gross Gamma LND716	4800-4420-89-095
FH112	Annulus - Elevation 305'	HPGe w/Ce-144/Eu-154	4240-3211-90-060



TABLE 2

SUMMARY OF AUXILIARY AND FUEL HANDLING BUILDING SNM INVENTORY

<u>SYSTEM</u>	<u>DETERMINED FUEL (UO<sub>2</sub>) QUANTITY</u>			<u>REFERENCE</u>
	<u>Measured Value</u>	<u>Uncertainty on Measured Values</u>	<u>MDL Value</u>	
Makeup & Purification	2.81 kg	+27%, -27%	0.60 kg	Appendix A
Liquid Radwaste Disposal	4.13 kg	+71%, -71%	0.11 kg	Appendix B
Solids Radwaste Disposal	0.01 kg	+57%, -57%	-	Appendix C
Balance of Systems	3.80 kg	+34%, -92%	-	Appendix D
Other areas	<u>0.00*</u> kg	<u>-</u>	<u>-</u>	Appendix E
Subtotal	10.75 kg		0.71 kg	
* * * * *	* * * * *	* * * * *	* * * * *	* * * * *

Total UO<sub>2</sub> Range: Low Value: 4.22 kg, High Value: 15.78 kg

Estimate of Record: 11.46 kg UO<sub>2</sub>

\* Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 kg UO<sub>2</sub>.

TABLE 3

## AREAS NOT REQUIRING SNM ASSESSMENT

<u>DESIGNATION</u>	<u>NAME</u>	<u>EXPLANATION</u>
AX002	Access Corridor	No water piping in area
AX003	Access Area	No water piping in area
AX022	North Stairwell	No water piping in area
AX023	Elevator Shaft	No water piping in area
AX027	South Stairwell	No water piping in area
AX103	MCC 2-11 EB	No water piping in area
AX104	MCC 2-21 EB	No water piping in area
AX105	Substation 2-11 E	No water piping in area
AX106	Substation 2-21 E	No water piping in area
AX107	MCC 2-11 EA	No water piping in area
AX108	MCC 2-21 EA	No water piping in area
AX121	Elevator Shaft	No water piping in area
AX122	North Stairwell	No water piping in area
AX123	Access Area	No water piping in area
AX133	South Stairwell	No water piping in area
AX135	Radwaste Disposal Control Panel	Same area as AX101 (see Appendix B)
AX201	North Stairwell	No water piping in area
AX202	Elevator Shaft	No water piping in area
AX203	4160 Switchgear 2-1 E	No water piping in area
AX204	4160 Switchgear 2-2 E	No water piping in area
AX205	RB Purge Air Supply	System design prevents fuel transport
AX206	RB Purge Exhaust - B	System design prevents fuel transport
AX207	RB Purge Exhaust - A	System design prevents fuel transport
AX208	AB Exhaust Unit B	System design prevents fuel transport
AX209	AB Exhaust Unit A	System design prevents fuel transport
AX210	FHB Exhaust Unit B	System design prevents fuel transport
AX211	FHB Exhaust Unit A	System design prevents fuel transport
AX213	Unit Substation	No water piping in area
AX215	FHB Supply Unit	System design prevents fuel transport
AX216	AB Supply Unit	System design prevents fuel transport
AX217	Access Area	No water piping in area
AX222	South Stairwell	No water piping in area
AX223	Air Handling Units	System design prevents fuel transport
AX301	Elevator Shaft	No water piping in area
AX302	North Stairwell	No water piping in area
AX303	Elevator and Stairwell Access	No water piping in area
AX401	Roof	No water piping in area
AX403	Damper Room	System design prevents fuel transport
FH013	Oil Drum Storage	No water piping in area
FH102	East Corridor	No water piping in area
FH104	West Corridor	No water piping in area
FH201	East Corridor	No water piping in area
FH202	West Corridor	No water piping in area

TABLE 4

SUMMARY OF MAKEUP AND PURIFICATION SYSTEM SNM INVENTORY

<u>AREA</u>	<u>DESCRIPTION</u>	<u>DETERMINED FUEL QUANTITY (KG)</u>	<u>UNCERTAINTY</u>	<u>CALCULATION/ REFERENCE</u>
AX004	Seal Injection Valve Room	0.03	± 58%	4800-3211-89-103, Rev. 0
AX005	Makeup Pump - 1C	0.00	-	4249-3232-91-001, Rev. 0
AX006	Makeup Pump - 1B	0.07	MDL	4550-3211-88-009, Rev. 0
AX007	Makeup Pump - 1A	0.23	MDL	4550-3211-87-027, Rev. 1
AX026	Seal Injection Filters	0.00	MDL	4550-3211-87-017, Rev. 1
AX112	Seal Return Coolers	0.30	MDL	4550-3211-87-025, Rev. 1
AX114	MU&P Demineralizer - 1A	1.06	± 44%	4249-3211-90-069, Rev. 0
AX115	MU&P Demineralizer - 1B	0.13	± 25%	4249-3211-90-069, Rev. 0
AX116	Makeup Tank	0.31	± 95%	4550-3211-87-038, Rev. 1
AX117	MU&P Filters	0.06	± 68%	4200-3233-90-053, Rev. 0
FH001	MU Suction Valves	0.46	± 85%	4200-3233-90-056, Rev. 0
FH003a	MU Discharge Valves	0.01	± 100%	4200-3233-90-057, Rev. 0
FH003b	MU Discharge Valves	0.10	± 100%	4200-3233-90-057, Rev. 0
FH002	Access Corridor			
FH004	W Valve Room	0.16*	± 50%	PDSR Appendix A
FH014	Annulus - Elevation 281'			
FH101	MU&P Valves	0.32	± 71%	4200-3233-90-055, Rev. 0
FH112	Annulus - Elevation 305'	Note 1		4200-3211-90-060, Rev. 0
Misc.	Embedding Piping	0.17	± 95%	PDSR Appendix A

\* \* \* \* \*

	<u>Measured Values</u>	<u>Uncertainty on Measured Values</u>	<u>MDL Values</u>
Estimate of Record	2.81 Kg	(+ 27%, - 27%)	0.60 Kg

\*Fuel quantity was determined using measured data for areas with similar service.

NOTE 1 - Measured fuel quantity was included in the WDL System total.

NOTE 2 - Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO<sub>2</sub>.

NOTE 3 - See Appendix F for details of data reduction.

TABL

SUMMARY OF LIQUID RADWASTE DISPOSAL SYSTEM SNM INVENTORY

<u>AREAS</u>	<u>DESCRIPTION</u>	<u>DETERMINED FUEL QUANTITY (KG)</u>	<u>UNCERTAINTY</u>	<u>CALCULATION/ REFERENCE</u>
AX011	AB Sump Tank Pumps & Valves	0.00	-	4200-3232-90-048, Rev. 1
AX012	AB Sump, Pumps & Tank	0.10	± 104%	4200-3232-90-048, Rev. 1
AX013	Evaporator Cond. Tanks & Pumps	0.00*	-	PDSR Appendix B
AX014	RC Evaporator	0.00	MDL	Reference 17
AX015a/b	Cleanup Filters	0.10	MDL	4249-3232-90-070, Rev. 0
AX016	Cleanup Demineralizer - 2A	0.00	MDL	4249-3232-90-070, Rev. 0
AX017	Cleanup Demineralizer - 2B	0.00	MDL	4249-3232-90-070, Rev. 0
AX018	Waste Transfer Pumps	0.00	MDL	4249-3232-90-071, Rev. 0
AX019	WDL Valve Room	0.01	MDL	4200-3232-90-049, Rev. 1
AX020	RC Bleed Holdup Tanks - B & C	3.50	± 83%	4800-3232-89-099, Rev. 0
AX021	RC Bleed Holdup Tank - A	0.31	± 46%	4800-3232-89-077, Rev. 0
AX024	AB Sump Filters	0.02	± 36%	4200-3232-90-064, Rev. 0
AX101	Radwaste Disp. Control Panel	0.00	-	PDSR Appendix B
AX102	RB Sump Pump Filters			PDSR Appendix B
AX131	Miscellaneous Waste Holdup Tank	0.10*	± 104%	PDSR Appendix B
AX134	Misc. Waste Holdup Tank Pump			PDSR Appendix B
AX128	Instrument & Valve Room	0.01*	± 95%	PDSR Appendix B
AX129	Deborating Demineralizer - 1B	0.00	-	4550-3232-87-016, Rev. 0
AX130	Deborating Demineralizer - 1A	0.00	-	4550-3232-87-016, Rev. 0
AX501	RB Spray Pump - 1A	0.01	± 76%	4249-3232-90-072, Rev. 0
AX502	RB Spray Pump - 1B	0.01	± 59%	4249-3232-90-072, Rev. 0
AX503	DHR Cooler & Pump - 1A	0.01	± 100%	4249-3232-90-072, Rev. 0
AX504	DHR Cooler & Pump - 1B	0.01	± 92%	4249-3232-90-072, Rev. 0
FH008	Neutralizer Tank Pump	0.00	-	4200-3233-90-054, Rev. 0
FH009	Neutralizer Tanks	0.00	-	4200-3233-90-054, Rev. 0
FH012	Neutralizer Tank Filters	0.00*	-	PDSR Appendix B
FH101	MU&P Valve Room	Note 1		4200-3233-90-055, Rev. 0
FH112	Annulus - Elevation 305'	0.01	± 55%	4240-3211-90-060, Rev. 0
Misc.	Embedding Piping	0.04	± 95%	PDSR Appendix B

\* \* \* \* \*

	<u>Measured Values</u>	<u>Uncertainty on Measured Values</u>	<u>MDL Values</u>
Estimate of Record	4.13 Kg	(+ 71%, - 71%)	0.11 Kg

\*Fuel quantity was determined using measured data for areas with similar service.

NOTE 1 - Fuel quantity was included in MU&P System total.

NOTE 2 - Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO<sub>2</sub>.

NOTE 3 - See Appendix F for details of data reduction.

TABLE 6

SUMMARY OF SOLIDS RADWASTE DISPOSAL SYSTEM SNM INVENTORY

<u>AREA</u>	<u>DESCRIPTION</u>	<u>DETERMINED FUEL QUANTITY (KG)</u>	<u>UNCERTAINTY</u>	<u>CALCULATION/ REFERENCE</u>
AX008	SRST (WDS-T-1B)	**		PDSR Appendix C
AX009	SRST (WDS-T-1A)	**		PDSR Appendix C
AX010	Spent Resin Transfer Pump	**		PDSR Appendix C
AX124	Conc. Liquid Waste Pump	0.00*	-	4200-3233-90-050, Rev. 0
AX218	Conc. Waste Tank (WDS-T-2)	0.01*	± 57%	4240-3233-90-061, Rev. 0
FH010	Reclaimed Boric Acid Tank	0.00	-	PDSR Appendix C
FH011	Reclaimed Boric Acid Pump	0.00	-	PDSR Appendix C
Misc.	Embedded Piping	0.00	-	PDSR Appendix C
* * * * *				
		Measured Values	Uncertainty on Measured Values	MDL Values
	Estimate of Record	0.01 Kg	(+ 57%, - 57%)	0 Kg

\*AX124 AND AX218 FUEL QUANTITY WAS DETERMINED BY ACTUAL MEASUREMENTS IN BOTH AREAS.

\*\*SNM INVENTORY WAS INCLUDED IN THE DEMINERALIZER VALUES, SEE APPENDIX C.

NOTE: Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO<sub>2</sub>.

TABLE 7

SUMMARY OF BALANCE OF SYSTEMS CONTAINING SNM INVENTORY

AREA	DESCRIPTION	PRIMARY SYSTEM	DETERMINED FUEL QUANTITY (KG)	UNCERTAINTY	CALCULATION/REFERENCE
AX113	Waste Gas Analyzer Room	WDG	0.00	-	PDSR Appendix D
AX118	Spent Fuel Cooler Area	SFC	0.00	-	PDSR Appendix D
AX119	Spent Fuel Demin. K-1	SFC	0.00	-	PDSR Appendix D
AX120	Spent Fuel Filters F-1A/1B	SFC	0.00	-	PDSR Appendix D
AX125	Waste Gas Decay Tank - 1B	WDG	0.00	-	PDSR Appendix D
AX126	Waste Gas Filter F-1	WDG	0.00	-	PDSR Appendix D
AX127	Waste Gas Decay Tank - 1A	WDG	0.00	-	PDSR Appendix D
FH105	Model Room	SNS	0.00*	-	4200-3551-90-051
FH106	Monitor Tanks & Sample Sink	SDS	0.00*	-	4200-3551-90-051
FH109	Spent Fuel Pool A	SFC	3.80*	+34%, -92%	4800-4420-89-095
FH110	Spent Fuel Pool B	SDS	0.00	-	PDSR Appendix D
FH111	Fuel Cast Storage	SFC	0.00	-	PDSR Appendix D
FH205	Annulus - Elevation 328'	DWCS	0.00	-	PDSR Appendix D
FH302	SDS Operating Area	SDS	0.00	-	PDSR Appendix D
FH303	New Fuel Storage Area	DWCS	0.00	-	PDSR Appendix D
FH304	Annulus - Above 347'6"	Electrical	0.00	-	PDSR Appendix D
FH305	Spent Fuel Pool Access Area	DWCS	0.00	-	PDSR Appendix D

\* \* \* \* \*

Estimate of Record	Measured Values	Uncertainty on Measured Values	MDL Values
	3.80 Kg	+34%, -92%	0 Kg

\*Areas were measured for fuel using gross gamma directional surveys.  
 NOTE: Quantities of fuel indicated as 0.00 can be assumed to contain less than 0.005 Kg UO<sub>2</sub>.

FIGURE 1

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 280'6" & 258'6"

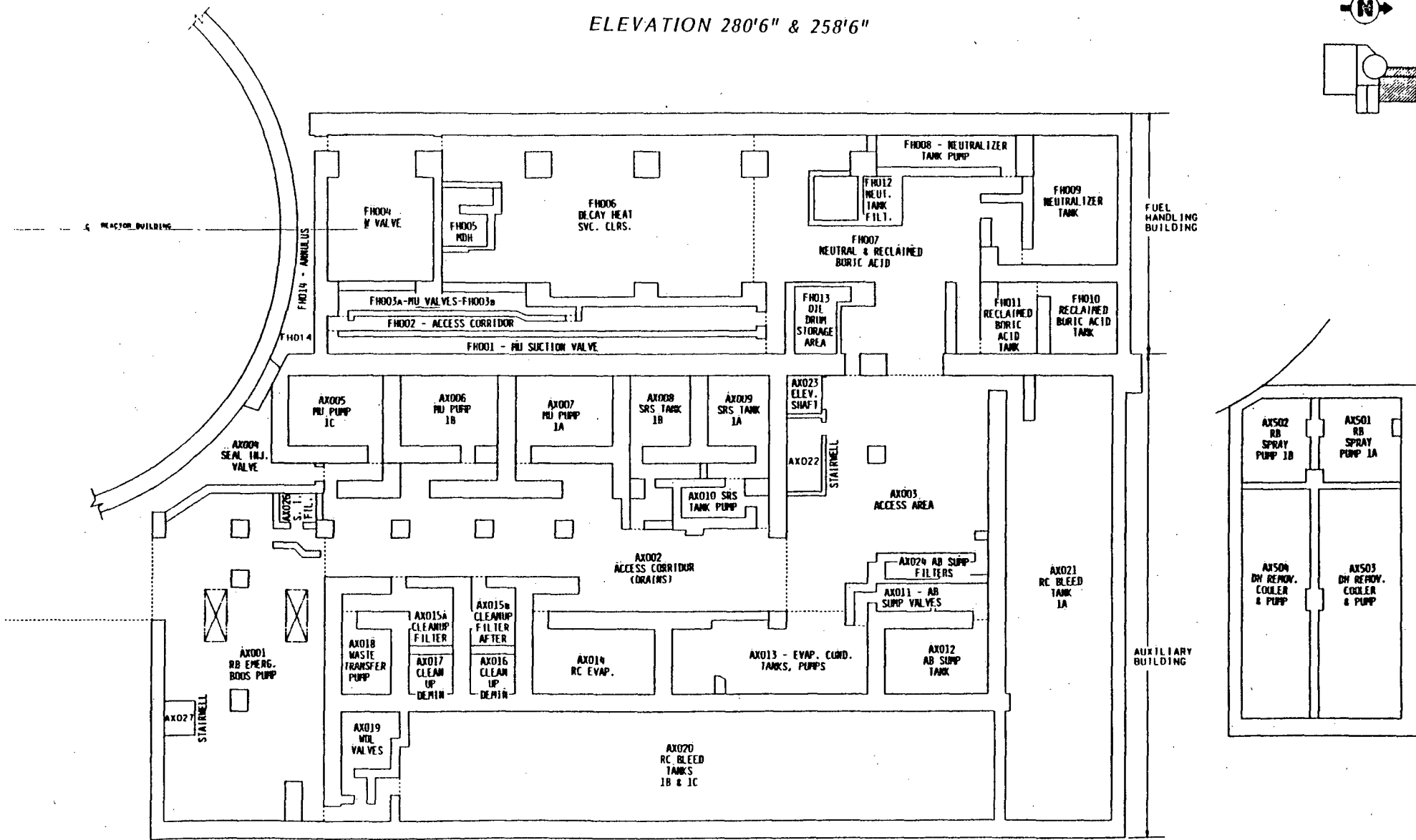
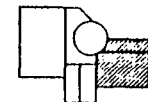
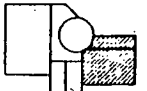


FIGURE 2

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 305'



KEY PLAN

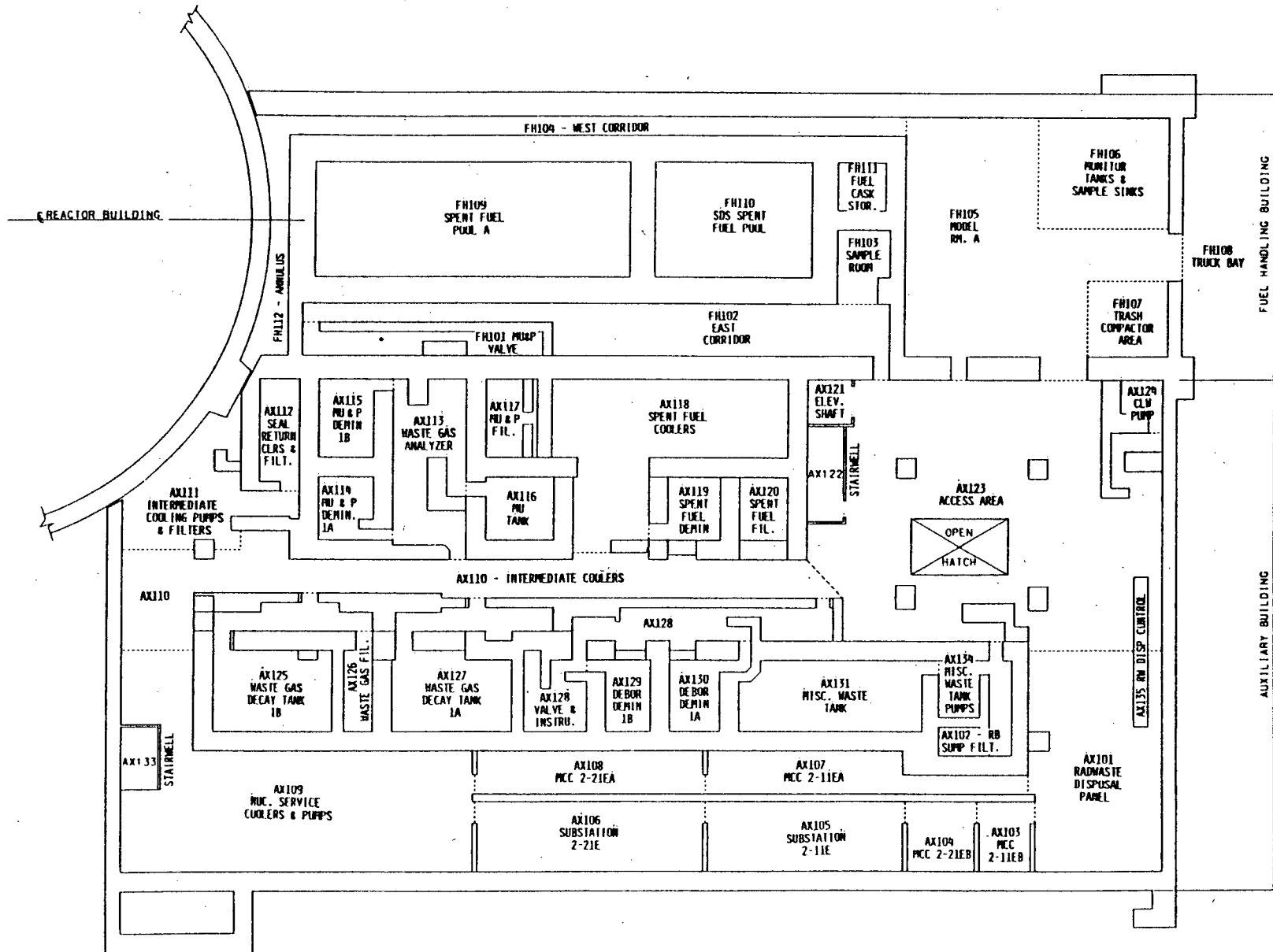




FIGURE 3

AUXILIARY AND FUEL HANDLING BUILDINGS

ELEVATION 328"

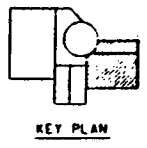
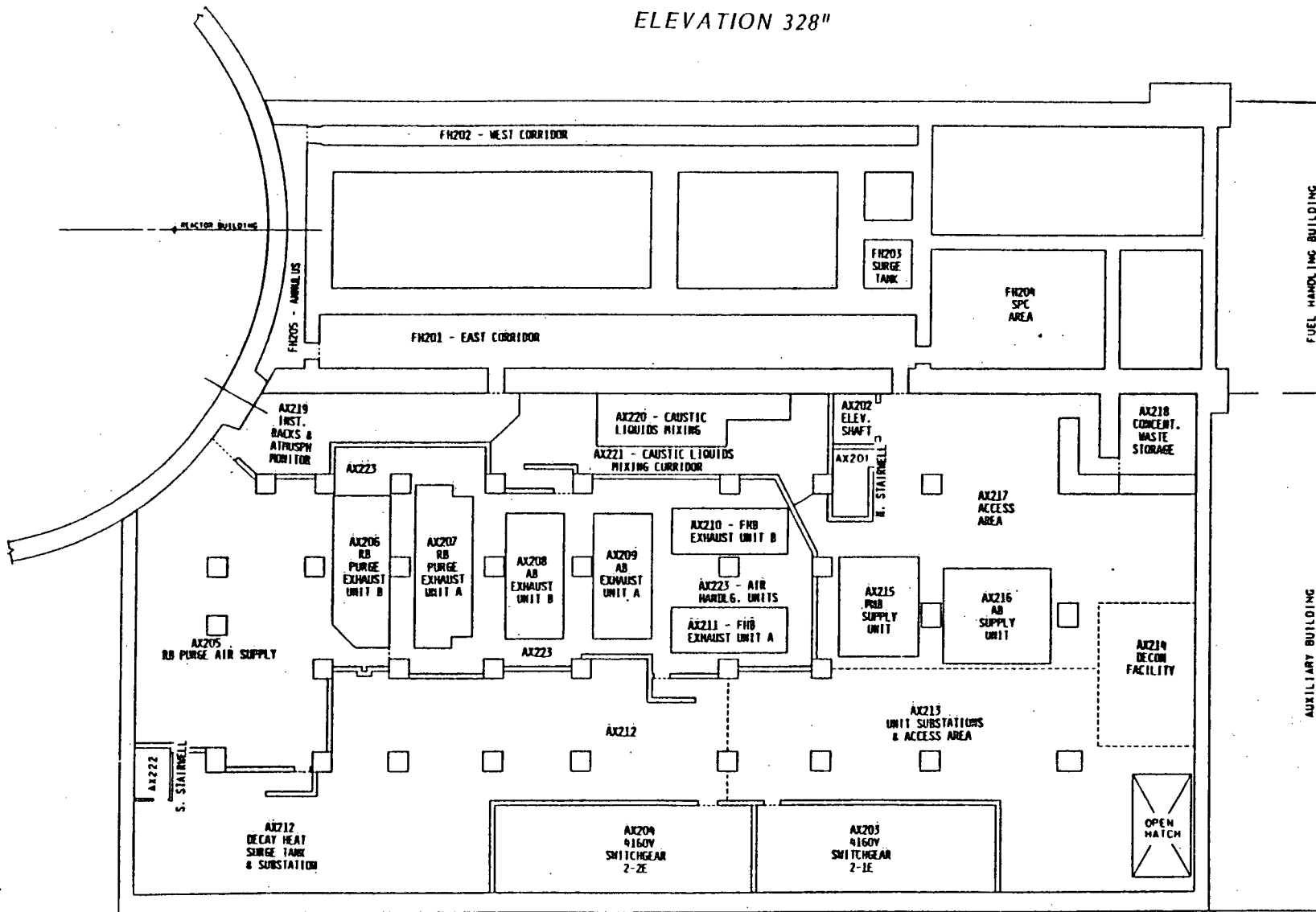
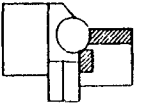


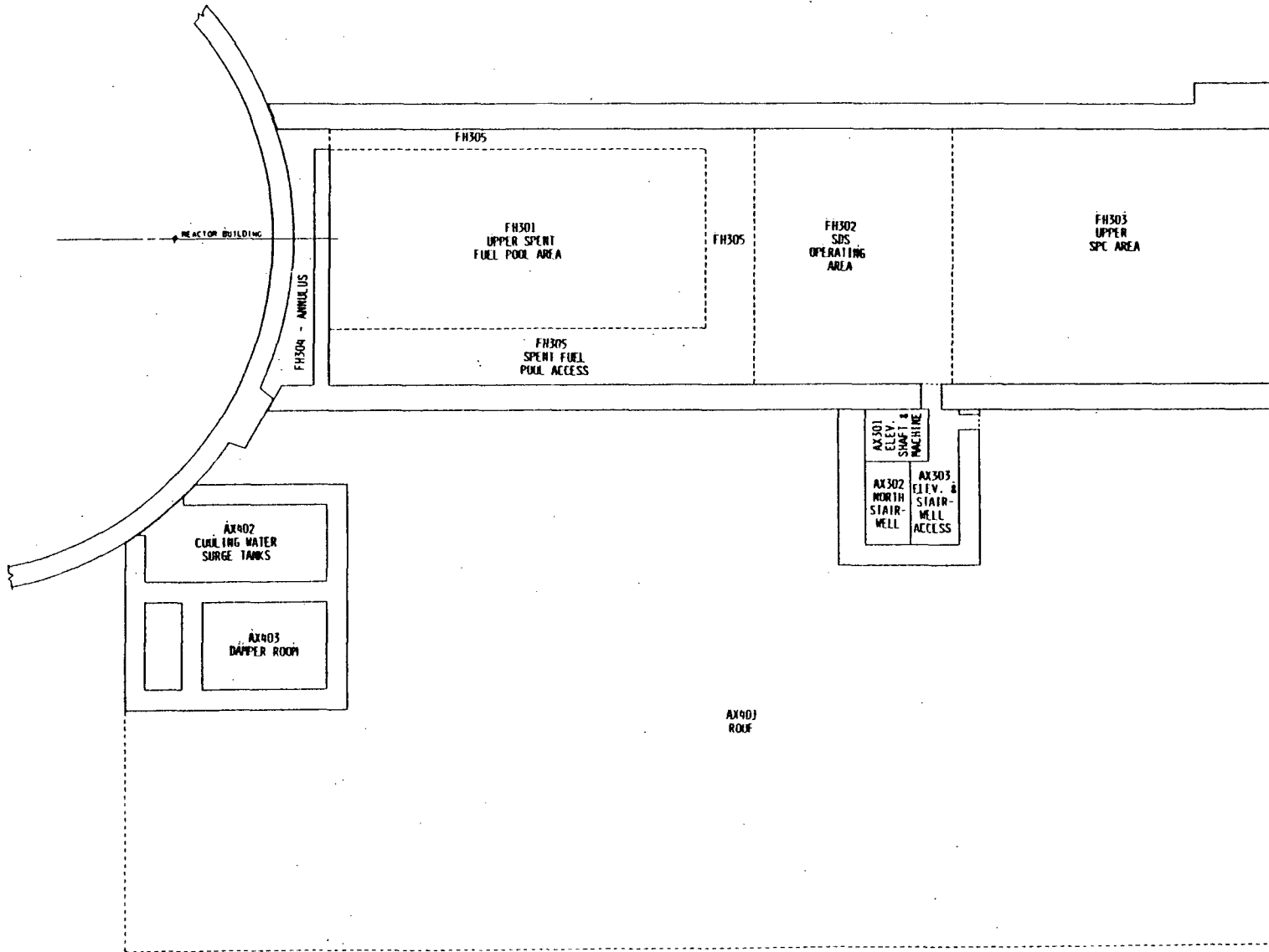
FIGURE 4

AUXILIARY AND FUEL HANDLING BUILDINGS

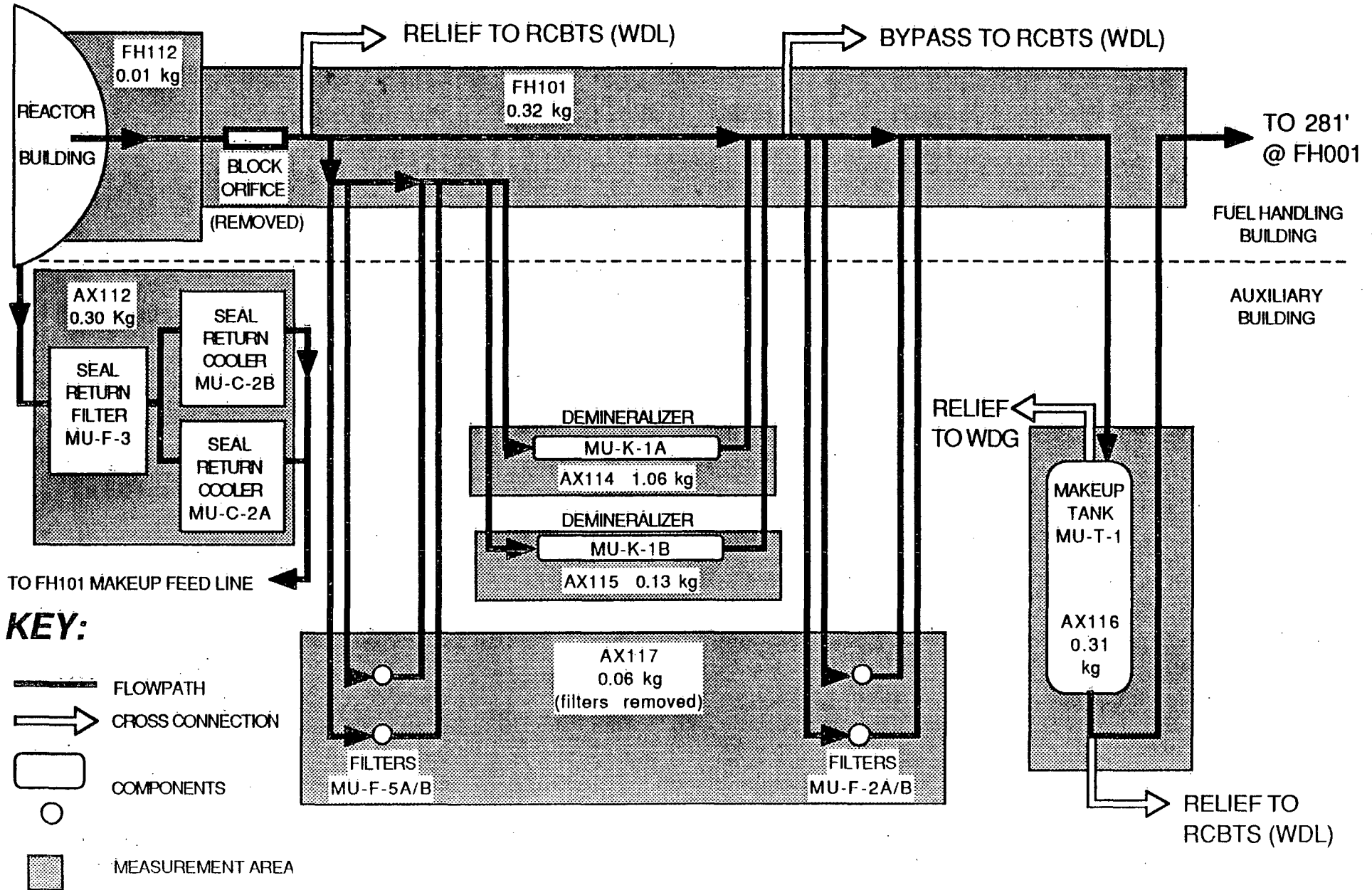
ELEVATION 347'6"



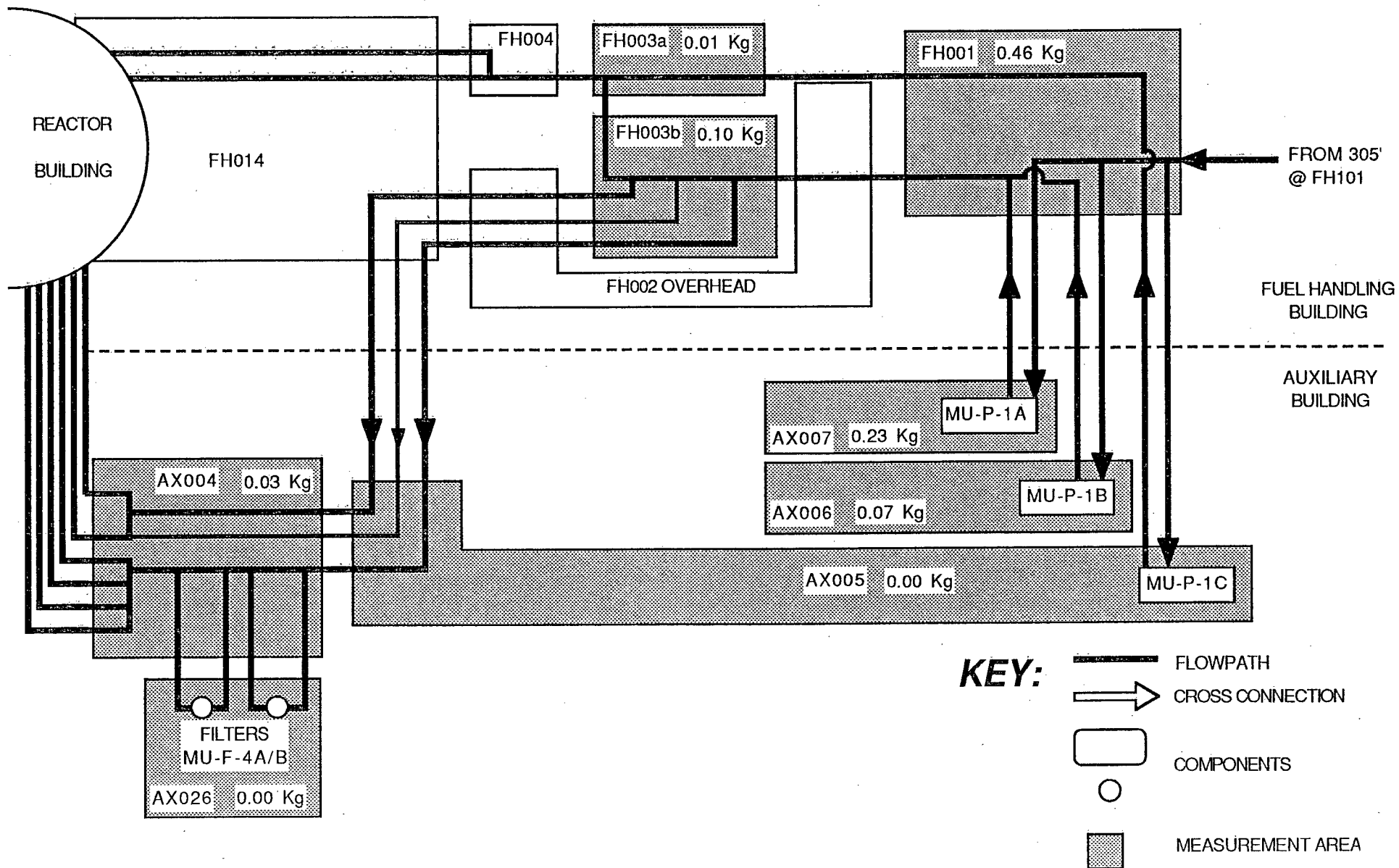
KEY PLAN



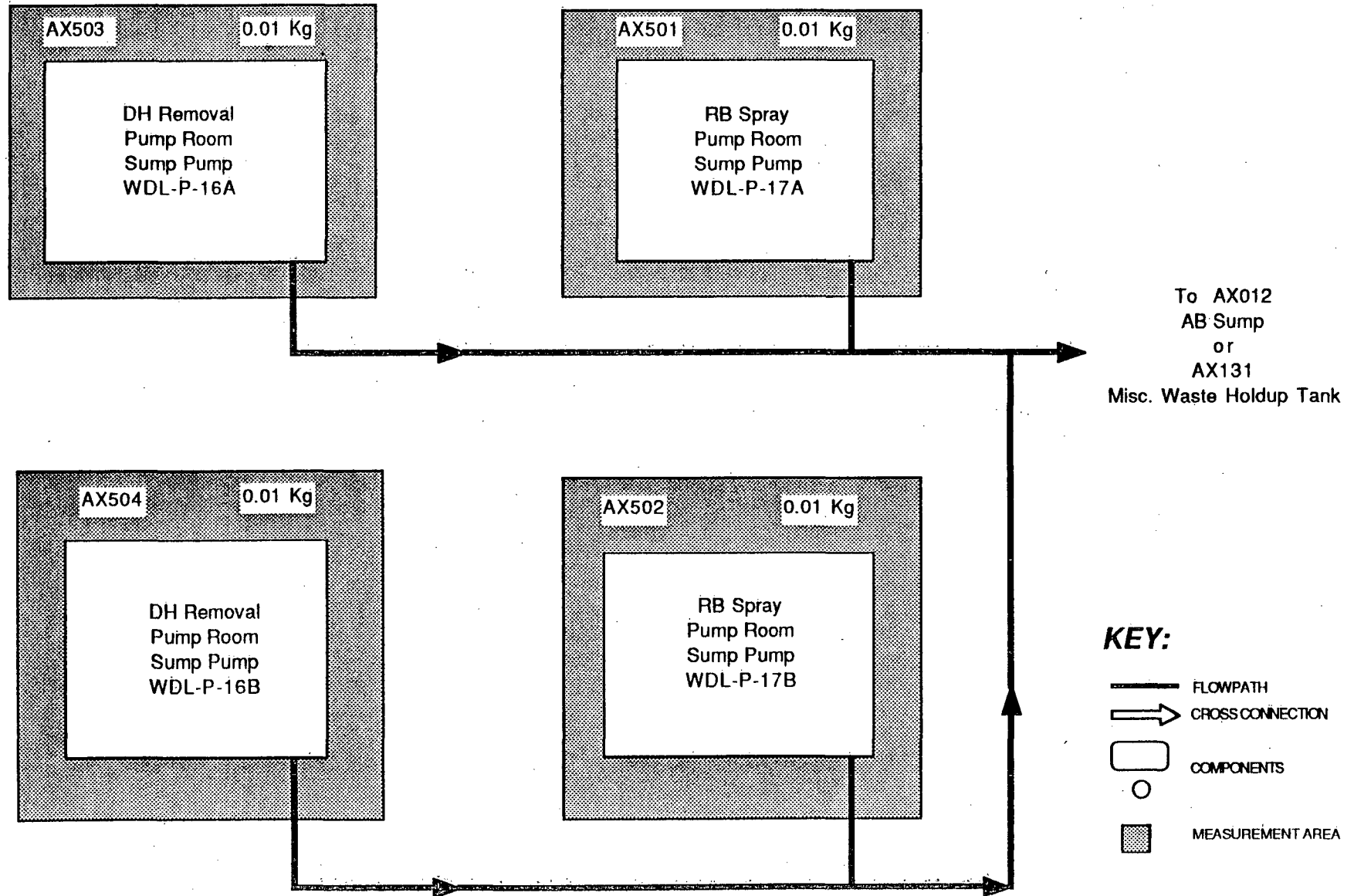
**FIGURE 5**  
**MAKEUP AND PURIFICATION SYSTEM**  
**FLOWPATH: FIRST SECTION, 305'-0" EL AFHB**



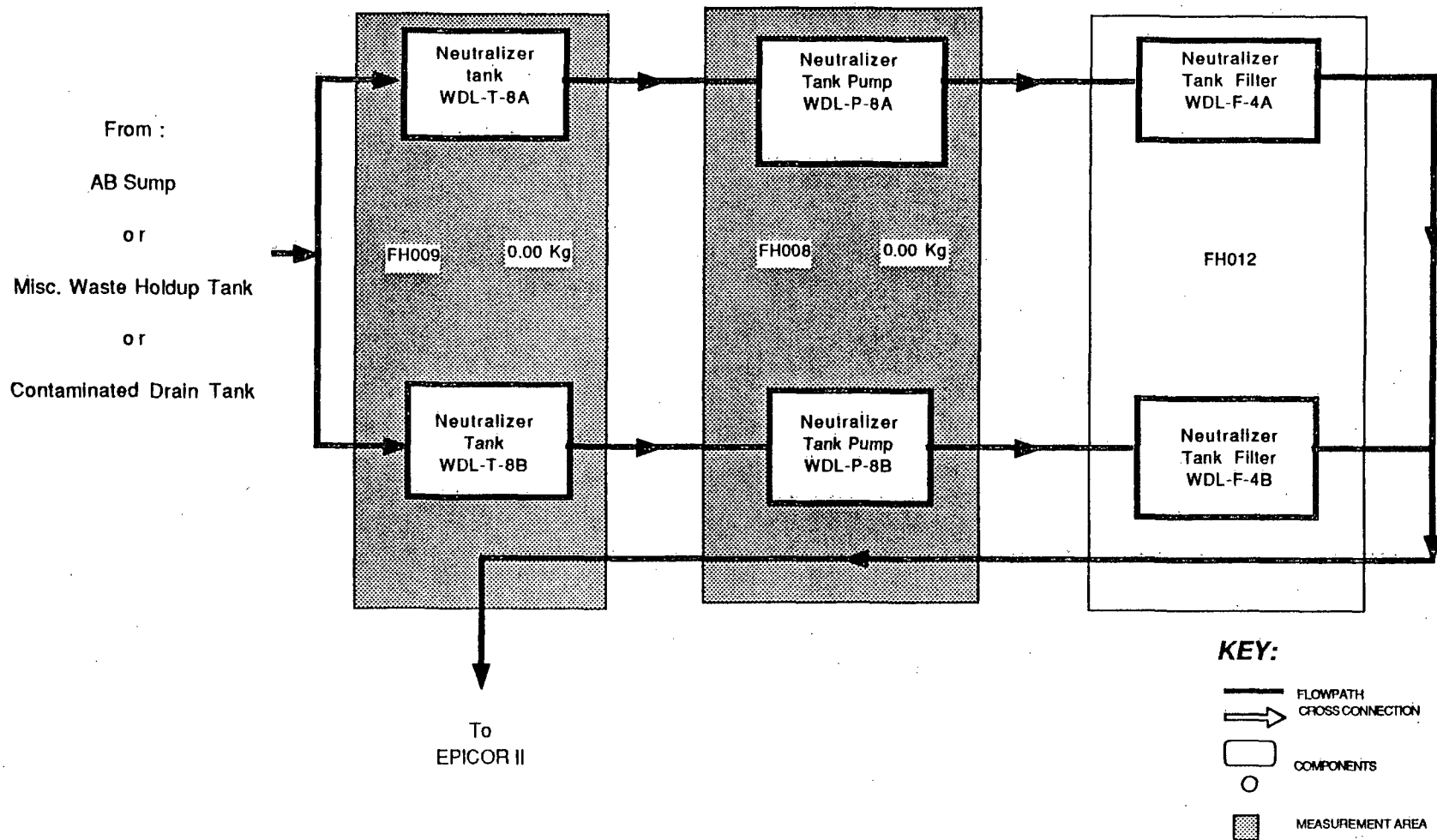
**FIGURE 6**  
**MAKEUP AND PURIFICATION SYSTEM**  
**FLOWPATH: SECOND SECTION, 281'-6" EL AFHB**



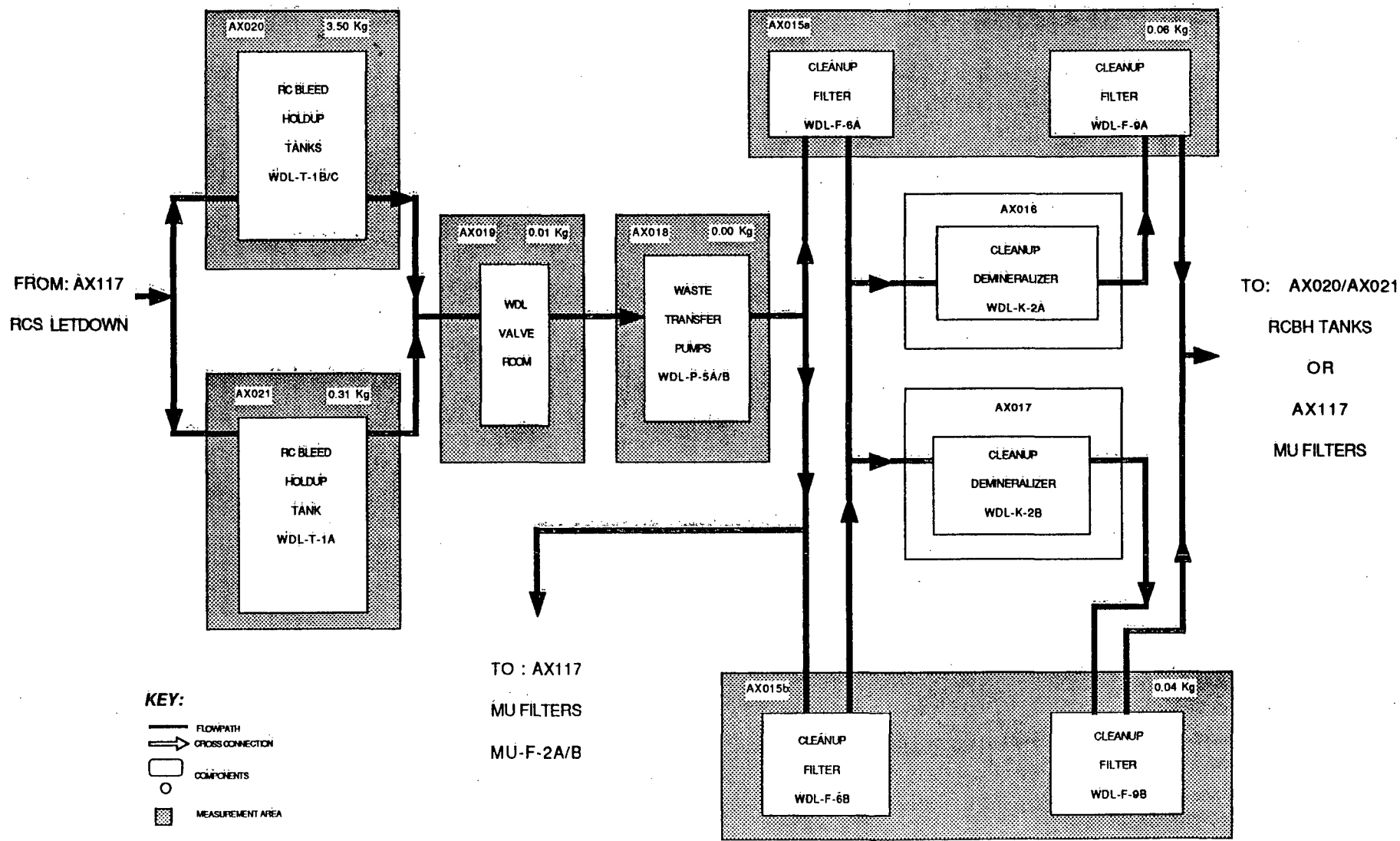
**FIGURE 7**  
**LIQUID RADWASTE DISPOSAL SYSTEM**  
**AUXILIARY BUILDING - ELEVATION 258' 6"**



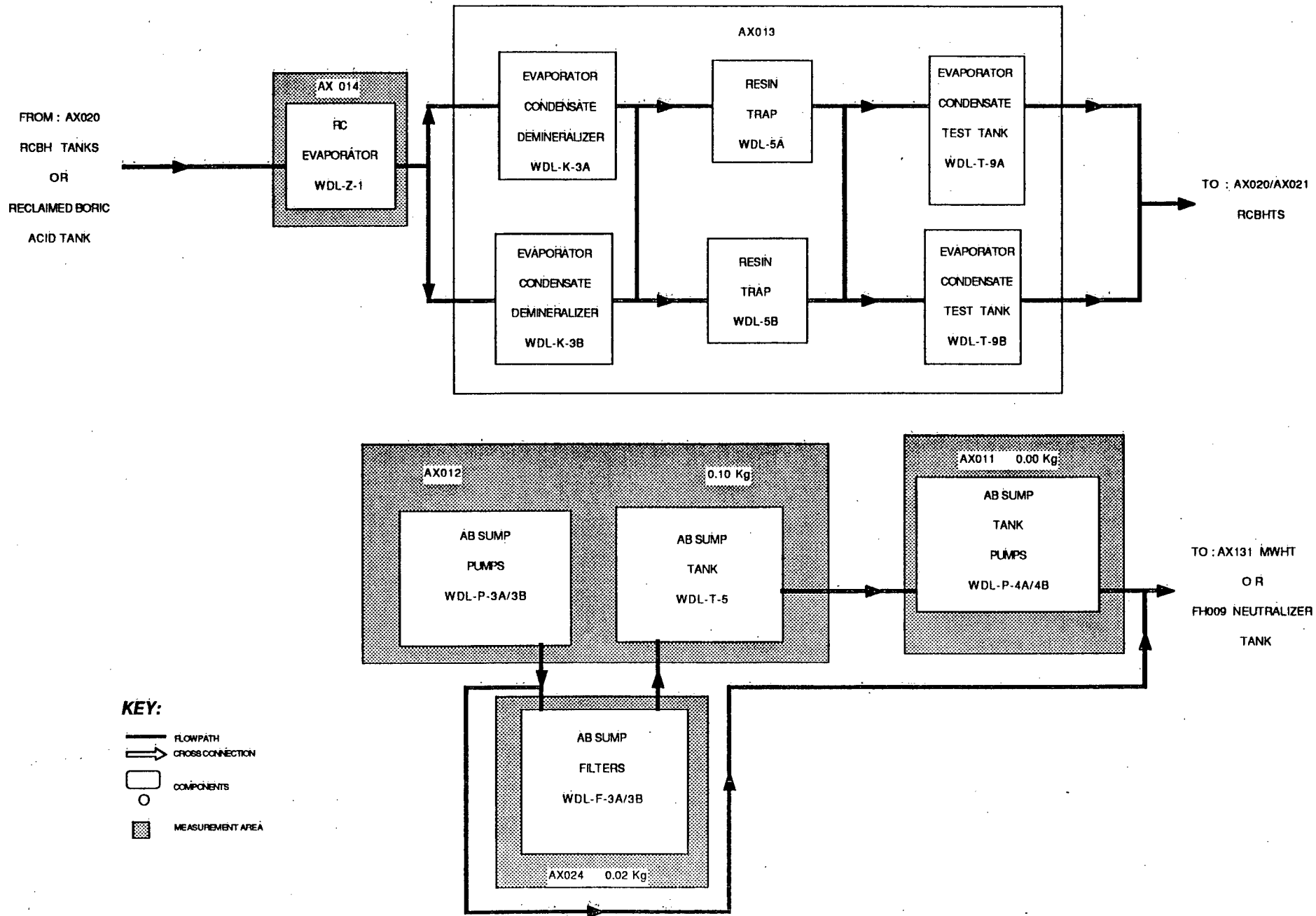
**FIGURE 8**  
**LIQUID RADWASTE DISPOSAL SYSTEM**  
**FUEL HANDLING BUILDING - ELEVATION 280' 6"**



**FIGURE 9**  
**LIQUID RADWASTE DISPOSAL SYSTEM - CLEANUP SUBSYSTEM**  
**AUXILIARY BUILDING - ELEVATION 280' 6"**



**FIGURE 10**  
**LIQUID RADWASTE DISPOSAL SYSTEM - RC EVAPORATOR AND AB SUMP SUBSYSTEMS**  
**AUXILIARY BUILDING - ELEVATION 280' 6"**

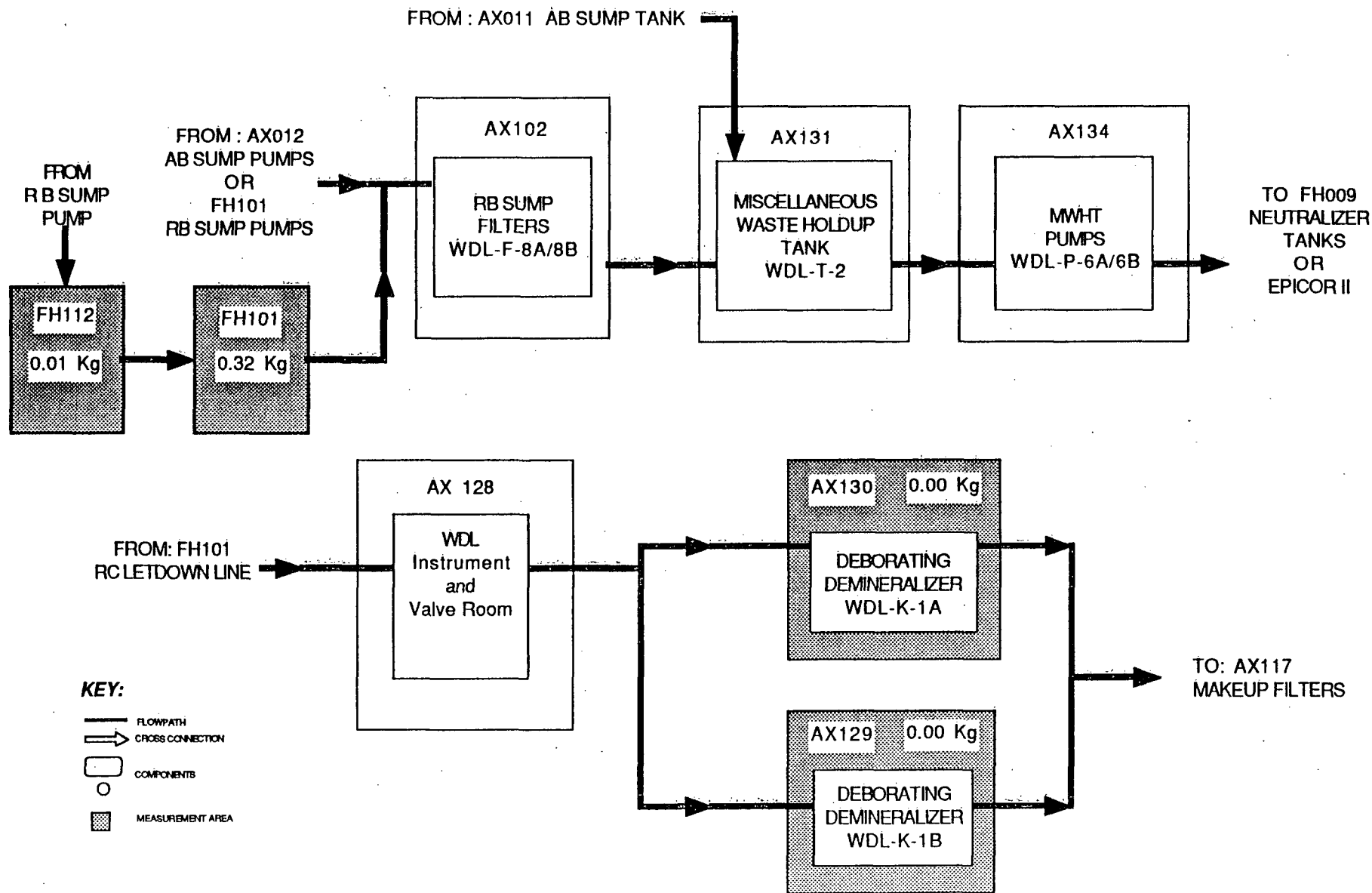


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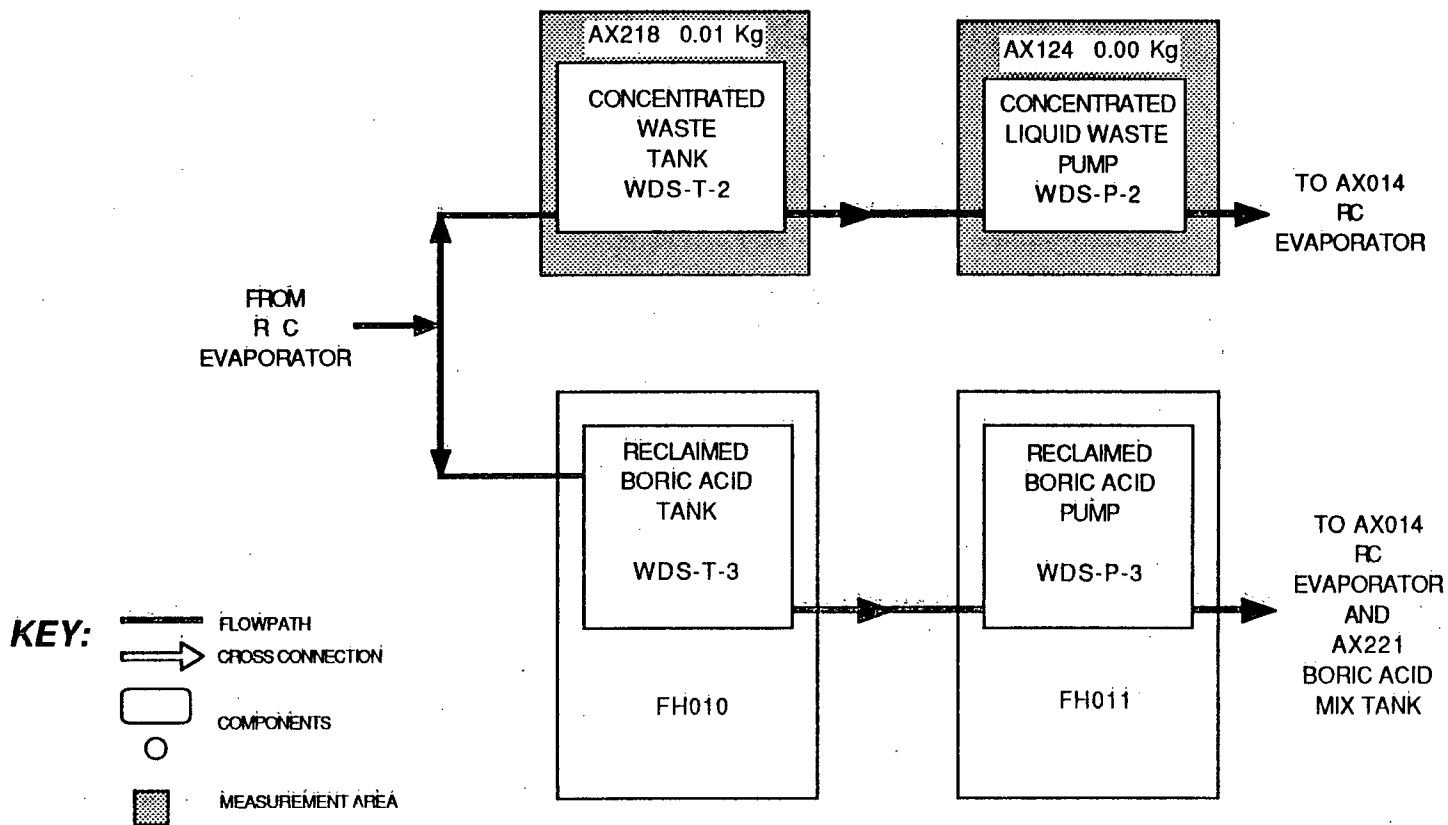
- FLOWPATH
- ⇨ CROSS CONNECTION
- COMPONENTS
- 
- ▨ MEASUREMENT AREA



**FIGURE 11**  
**LIQUID RADWASTE DISPOSAL SYSTEM - MISCELLANEOUS WASTE**  
**AUXILIARY BUILDING - ELEVATION 305'**



**FIGURE 12**  
**SOLID RADWASTE DISPOSAL SYSTEM -CONCENTRATED WASTE**  
**SUBSYSTEM FLOWPATH - AFHB - ELEVATIONS 281'6",305' & 328'**



**FIG E 13**  
**SOLID RADWASTE DISPOSAL SYSTEM - RESIN WASTE SUBSYSTEM**  
**FLOWPATH - AUXILIARY BUILDING - ELEVATION 281' 6"**

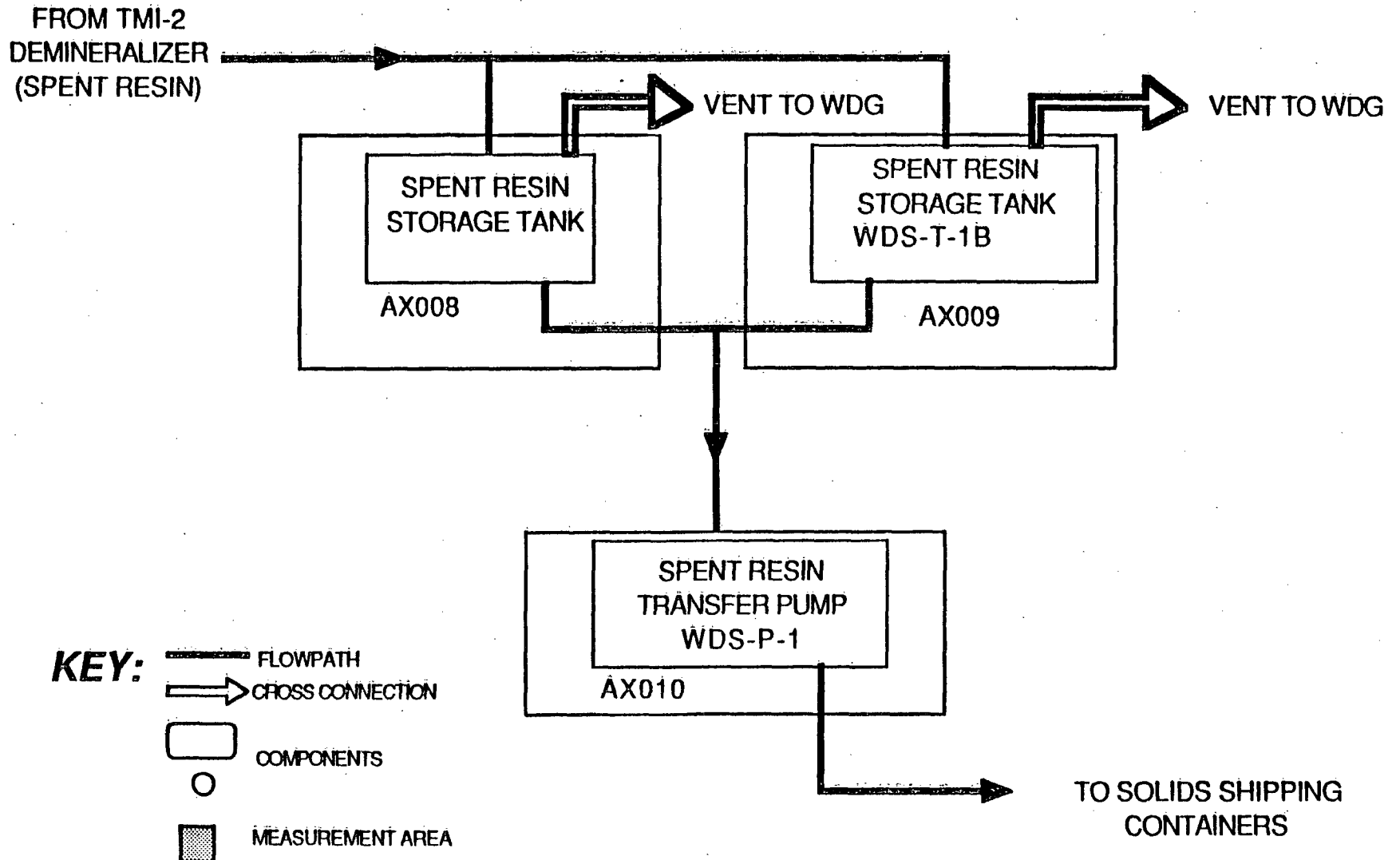
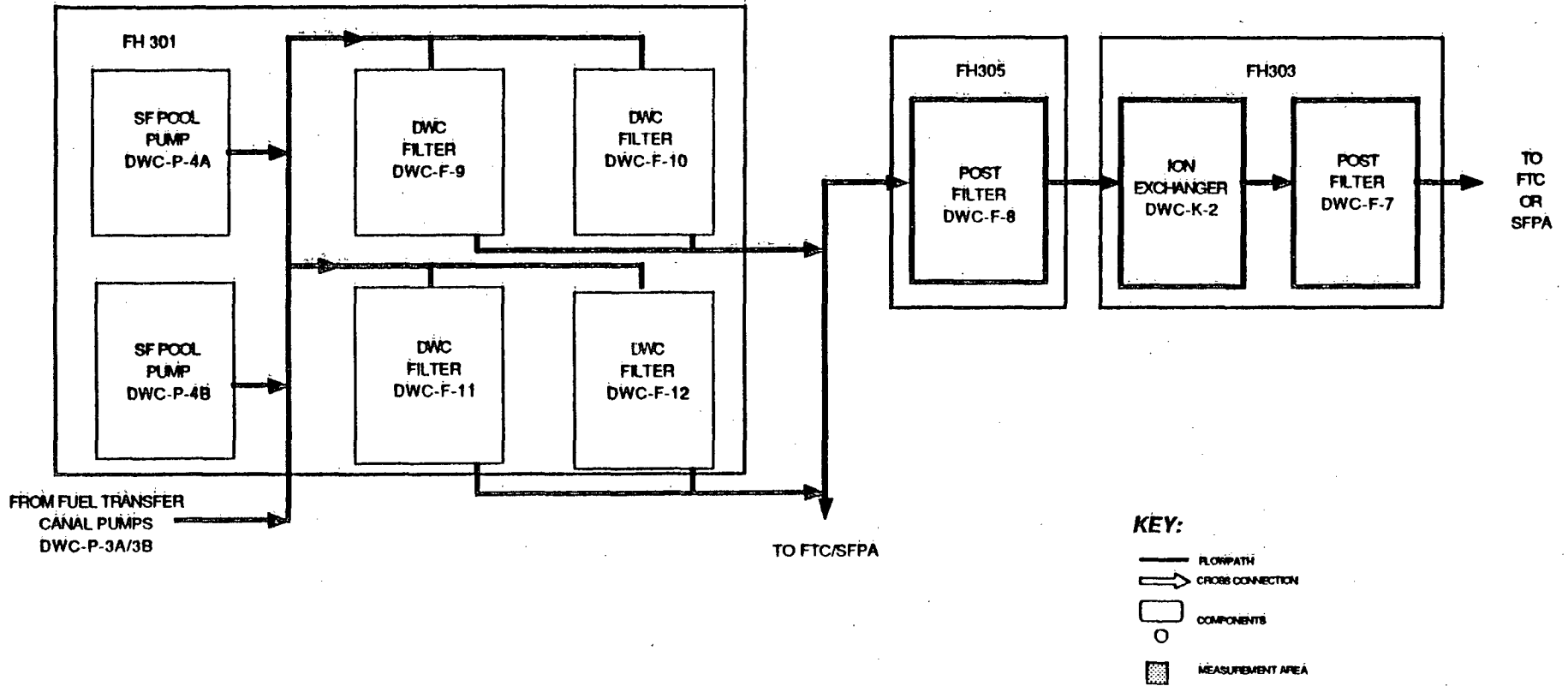
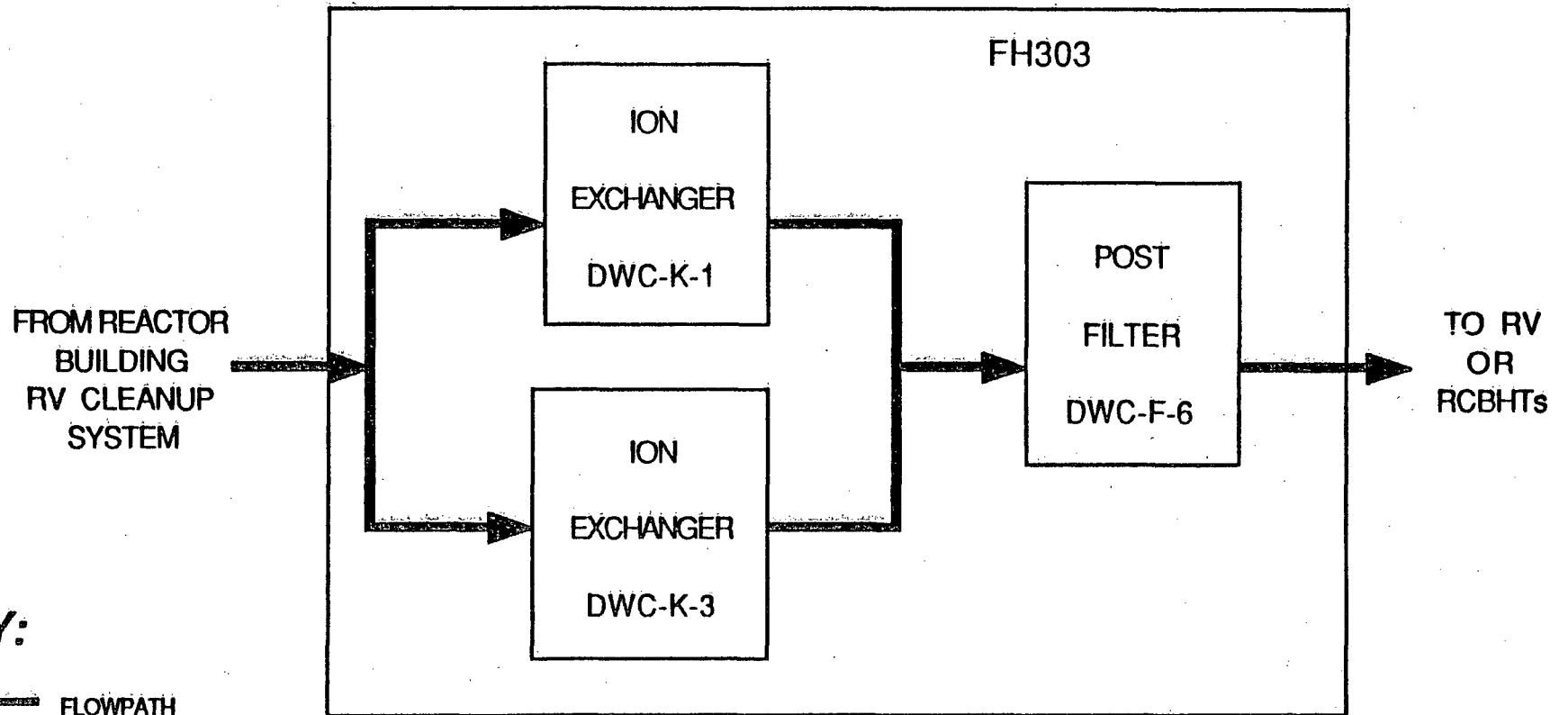


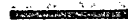
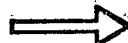



FIG. E. 14  
**DWC FUEL TRANSFER CANAL/SPENT FUEL POOL CLEANUP SYSTEM  
 FUEL HANDLING BUILDING - ELEVATION 347'6"**



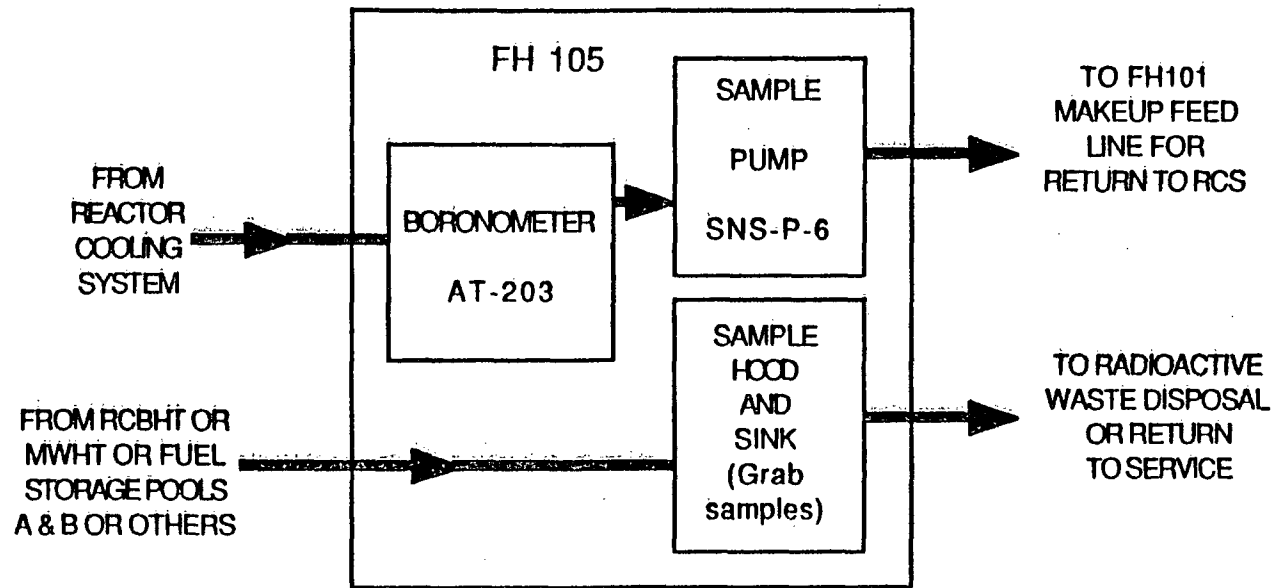
**FIGURE 15**  
**DWC REACTOR VESSEL CLEANUP SYSTEM**  
**FUEL HANDLING BUILDING - ELEVATION 347' 6"**




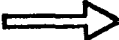
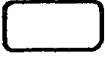


**KEY:**

-  FLOWPATH
-  CROSS CONNECTION
-  COMPONENTS
- 
-  MEASUREMENT AREA

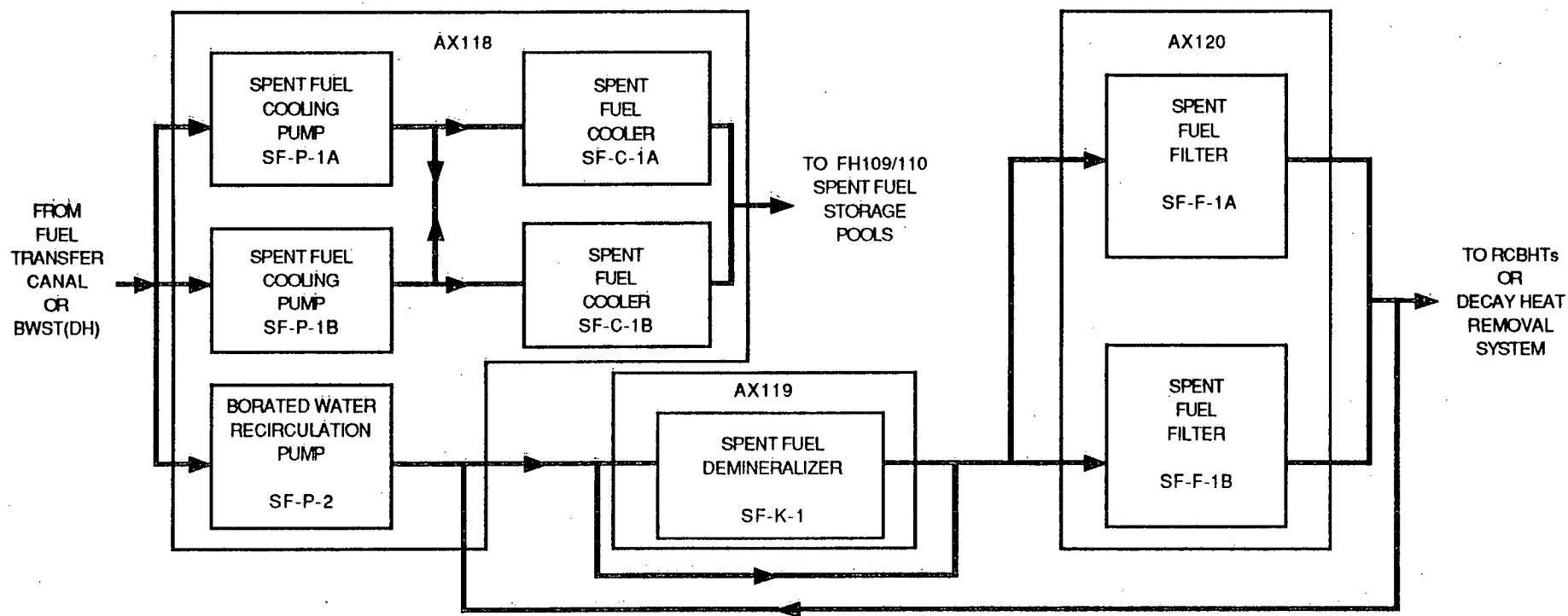
**FIGURE 16**  
**NUCLEAR SAMPLING SYSTEM**  
**FUEL HANDLING BUILDING - ELEVATION 305'**








**KEY:**

-  FLOWPATH
-  CROSS CONNECTION
-  COMPONENTS
- 
-  MEASUREMENT AREA

**FIGURE 17  
SPENT FUEL COOLING SYSTEM  
AUXILIARY BUILDING - ELEVATION 305'**

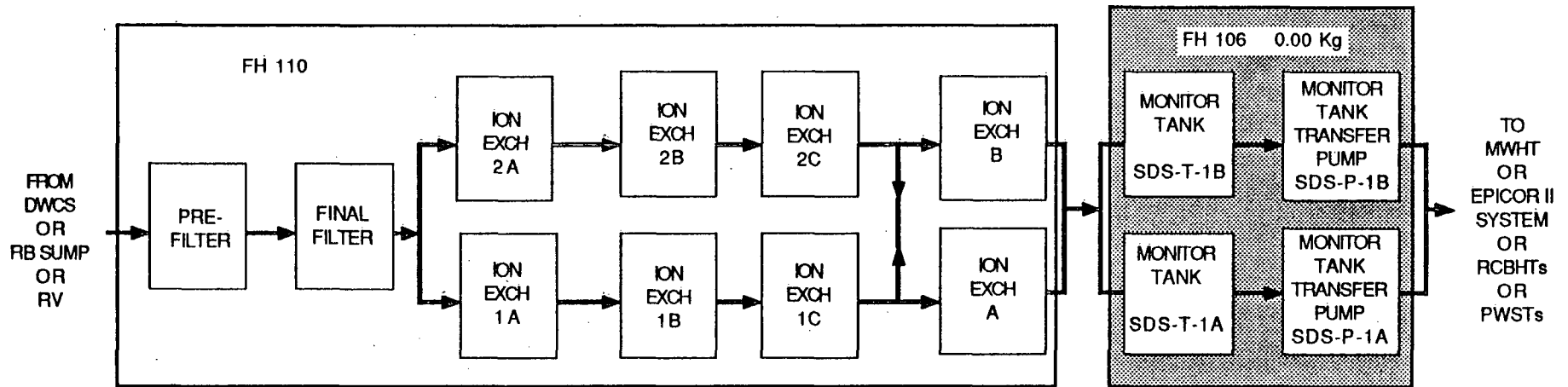


**KEY:**

-  FLOWPATH
-  CROSS CONNECTION
-  COMPONENTS
-  O
-  MEASUREMENT AREA

NOTE: AX118 PIPING ALSO USED DURING RECOVERY AS A FLOWPATH FOR THE DEFUELING WATER CLEANUP SYSTEM.

**FIGURE 18**  
**SUBMERGED DEMINERALIZER SYSTEM**  
**FUEL HANDLING BUILDING - ELEVATION 305'**

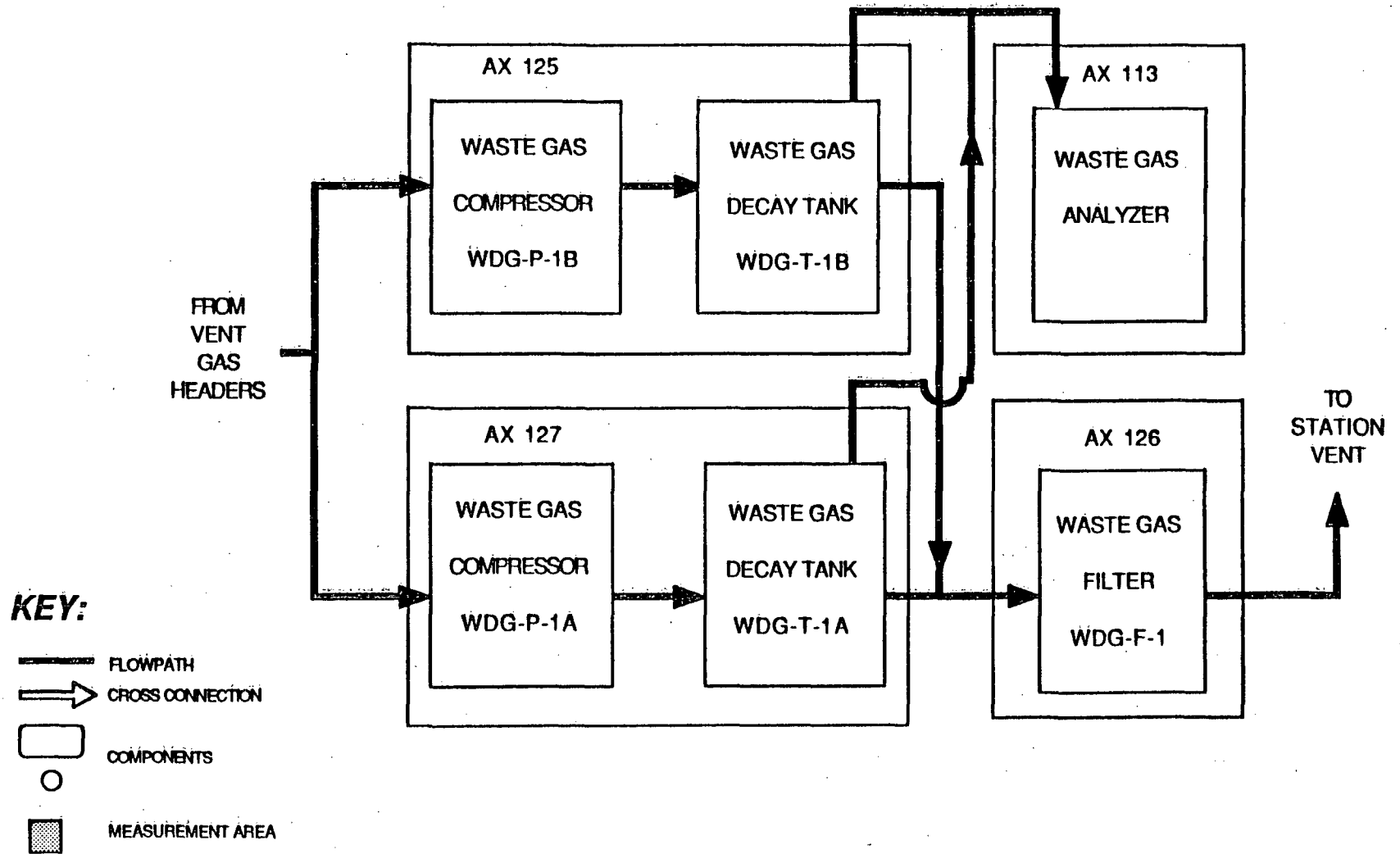


**KEY:**

- FLOWPATH
- CROSS CONNECTION
- COMPONENTS
- MEASUREMENT AREA



**FIGURE 19**  
**GAS RADWASTE DISPOSAL SYSTEM**  
**AUXILIARY BUILDING - ELEVATION 305'**



## APPENDIX A

### MAKEUP AND PURIFICATION SYSTEM

#### 1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel ( $UO_2$ ) remaining in the Makeup and Purification (MU&P) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 5 and 6 which shows the primary flowpaths between areas containing MU&P components and piping in the AFHB. All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the MU&P system, and the part played during the 1979 TMI-2 Accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing MU&P system components and piping. The majority of areas were measured using gamma spectroscopy featuring either shielded sodium iodide scintillation or high purity germanium detectors connected to multi-channel analyzers. Since the presence of Ce-144(Pr-144) was not always identifiable, a MDL calculation was prepared when necessary.

Section 4, "Analysis", explains how the estimate of record of the amount of fuel in the MU&P system was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the MU&P system and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analyses performed.

## 2.0 BACKGROUND

Prior to the accident, the Makeup and Purification (MU&P) System was designed to control the Reactor Coolant System (RCS) water inventory as well as maintaining the water quality and chemistry within specifications. The system also served to:

- a. provide seal injection water to the reactor coolant pumps
- b. provide a means to separate and vent gases from the coolant
- c. add makeup water to the Core Flood tanks
- d. provide High Pressure Injection (HPI) water to the RCS in the event of a Loss-of-Coolant Accident (LOCA).

The MU&P System is interconnected to the following auxiliary systems in order to perform its functions:

- a. Chemical Addition System (CA)
- b. Radioactive Waste Disposal Systems (WDS, WDL, WDG)
- c. Core Flood System (CF)
- d. Decay Heat Removal System (DH)
- e. Nitrogen Gas System (NM)
- f. Demineralized Water System (DW)

During normal plant operation, reactor coolant flow was confined to a closed loop from the reactor core through the steam generators and back to the reactor. In order to maintain water quality, it was necessary to pass

a portion of the flow (usually 35 gallons per minute) through a series of filters and demineralizers to remove contaminants. The water was then directed to the makeup tank (MU-T-1). The makeup tank served as a reservoir for coolant letdown from the RCS and provided a water source for the makeup pumps which return the water to the RCS. The makeup tank was also used to separate radioactive gases from the coolant. These gases were collected in the gas space of the MU tank and vented to the WDG System. The safety function of the MU&P System was to provide a high pressure flow of additional coolant to the core from the Borated Water Storage Tank in the event of a rupture in the RCS system piping.

During the TMI-2 accident, while the reactor coolant was being letdown from the RCS, fuel particles were distributed throughout the MU&P System. Part of the fuel deposited in the MU&P System crossed over into the interconnected systems during the defueling and recovery operations performed from 1979 through 1989. Additional fuel was relocated during the TMI-2 cleanup. Directional radiation surveys indicated that fuel and/or fission products were dispersed throughout much of the primary piping system as finely divided particles and/or as plating on surfaces. In general, the MU&P System piping and components have been flushed with processed water. In addition, the letdown block orifice and the Makeup filters have been removed. Several portions of the MU&P piping and components have been physically isolated and drained. A major portion of the remaining fuel deposited in the Makeup and Purification System is concentrated in large volume containers such as tanks and demineralizers and is mixed with sediment and demineralizer resins.

### 3.0 METHODS

Standard gamma spectroscopy methods utilizing NaI (TI) and HPGe detectors were employed to measure the remaining fuel ( $UO_2$ ) in 12 of the 18 areas

occupied by the Makeup and Purification System. Ce-144(Pr-144) was selected as a tracer for fuel because of its reasonable half-life (284 days), relative high abundance, chemical similarity to reactor fuel which results in a low escape rate coefficient from reactor fuel, and its 2.19 MeV gamma-ray that is readily identifiable on a multi-channel analyzer (MCA). The ratios of Ce-144-and Eu-154-to-fuel ( $UO_2$ ) were developed using sample data (Reference 6). Details of the specific methods used in each area are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. When a Ce-144(Eu-154) peak was not identifiable, a MDL calculation was performed in accordance with Reference 14. The gross counts in the region of interest (ROI) for Ce-144 (Eu-154), converted to an MDL value, was divided by the measured detector efficiency, the calculated photon fluence rate per kilogram of fuel for the geometry and count time under consideration.

One (1) area, AX005, utilized a series of exposure rate measurements made with tungsten-shielded directional probes to quantify the amount of residual fuel. The remaining two (2) measured areas, AX114 and AX115, were initially measured using gamma-ray spectrometry (Reference 15), but their residual fuel values were revised using improved sample data obtained in 1983.

Three (3) other areas, FH002, FH004 and FH014, containing MU&P valves and piping were not formally measured but were analyzed based on their service and measured fuel deposits in similar areas. Details of this determination are discussed in Section 4.0, Analysis, of this appendix.

#### 4.0 ANALYSIS

The MU&P System contains 0.4% of the fuel remaining at TMI-2. The MU&P System occupies portions of eighteen (18) areas in the Auxiliary and Fuel

Handling Buildings (AFHB), which are illustrated in Figures 5 and 6. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan, Reference 1. Fifteen (15) of these areas have been measured for fuel content, and the results are summarized in Table 4. Four (4) of the measured areas (AX006, AX007, AX026 and AX112) resulted in MDL values. Details of the methods and assumptions used to calculate the remaining fuel for each area can be found in the engineering calculations listed in Table 4.

The residual fuel determination for the MU&P Demineralizers "A" and "B" (AX114 and AX115) was based on resin/fuel samples taken in March and April 1983. These samples were believed to be more representative of the demineralizer radionuclide contents than previous samples. These samples were analyzed by Oak Ridge National Laboratory (ORNL), and the results (Reference 10) were used to determine the residual fuel values shown in Engineering Calculation 4249-3211-90-069, Revision 0 (Reference 16). The final fuel value for A Demineralizer was 1.06 Kg of  $UO_2$  and for B Demineralizer was 0.13 Kg of  $UO_2$ .

Most of the resins have been removed from the 'A' demineralizer and some resins have been removed from the 'B' demineralizer. Flush water does not flow well out of the 'A' demineralizer and it is assumed that the resin outlet piping is filled with a hard, packed resin/fuel mixture. The 'B' demineralizer still contains a hard, agglomerated resin/fuel mixture in the lower portion of the tank.

Since both MU&P Demineralizers were measured for fuel ( $UO_2$ ) content while full of resins, any fuel laden resins transferred out of the demineralizers should have reduced the quantity of fuel remaining in the MU&P Demineralizers. The measured residual fuel value for each demineralizer could be reduced to correct for any fuel transferred to the

Spent Resin Storage Tanks. However, since the amount of fuel transferred is unknown and radiation levels in both areas, AX114 and AX115, are still very high, it is assumed that most of the fuel originally measured is still in the piping at TMI-2. Some of the resin/fuel still remains in the 'A' and 'B' demineralizers (AX114 and AX115) and it is assumed that the balance of the fuel is deposited in the pipe lines between the demineralizers and the spent resin storage tanks and pump (AX008, AX009 and AX010). Since the measured residual fuel values for each MU&P Demineralizer are reliable data by themselves, it is assumed for purposes of SNM accountability that much of the fuel originally deposited in the MU&P Demineralizers is still remaining at TMI-2. Therefore, any fuel transferred from the demineralizers was already included in the residual fuel values established for the demineralizers and no additional fuel amounts are included for the Spent Resin Storage Tanks (AX008 and AX009) and the Spent Resin Storage Tank Pump (AX010).

Both of the MU&P demineralizers had been previously measured in 1982 using a collimated Si(Li) Compton recoil gamma-ray spectrometer (A Demineralizer) and a Be ( $\gamma,n$ ) detector (B Demineralizer) collecting gamma-ray flux data above 1.667 MeV. Even though the measured values were greater than the values obtained using the ORNL sample data, the measured values support the conclusion drawn in this PDSR. Details of the Be ( $\gamma,n$ ) and Si(Li) measurements can be found in Reference 15. The ORNL analyzed sample data was selected over the Si(Li) and Be ( $\gamma,n$ ) data for use in this PDSR because the sample analysis utilized a direct measurement of Uranium. The Si(Li) and Be ( $\gamma,n$ ) measurements relied on an analog ratio which inherently results in a larger uncertainty.

The three (3) remaining areas that were not formally measured and are the Makeup Valve Access Corridor (FH002), the Westinghouse (W) Valve Room (FH004), and the 281' Elevation Annulus (FH014). FH002 is an enclosed

corridor between the Makeup Suction and Makeup Discharge Valve Rooms (FH001 and FH003a/b), which has HPI, pressurizer level control, and reactor coolant pump seal injection lines running through the overheads above its concrete slab ceiling. All of these lines originate in the assayed area of FH003a/b, have been flushed several times since the accident, and are assumed to contain very little fuel. These lines also pass through the other unmeasured areas; with the two HPI lines running through both FH004 and FH014, and the remaining lines running through FH014 into AX004, which is an assayed area.

Since these unmeasured areas contain pipes and valves which were exposed to similar core debris and are physically similar in layout and design to those contained in the 305' elevation Makeup Valve Room (FH101), a conservative appraisal of the unmeasured segments can be made using the FH101 value of 0.320 kg of  $UO_2$ . First, the amount and size of valves and pipes in FH101 is significantly greater than the total amount contained in the three unmeasured areas. Second, FH101 contains letdown piping that carried primary coolant which had not been filtered prior to entering FH101.

It is very unlikely that the three (3) areas (FH002, FH004 and FH014) would contain exactly the same amount of fuel as FH101. It is more likely that the amount of fuel in FH101 is an upper bound value for the amount of fuel in the three (3) areas. This argument is supported by the most recent general radiation survey data for each of the four (4) areas. FH101 survey data indicates an average general area exposure rate of 470 mR/hr while FH002, FH004 and FH014 indicate an average general exposure rate of 43 mR/hr. FH101 contains all of the letdown pathways which is where most of the core debris was initially transported during the accident while the other three (3) areas contain makeup pathways which generally contains much less residual fuel. In view of these arguments,



it is assumed that the total quantity of residual fuel in FH002, FH004 and FH014 is  $0.16 \pm 0.08$  kg of  $UO_2$ .

Approximately 10% of the process piping was assumed to be embedded in concrete walls and floors, and was not included in the fuel determination for the measured areas. Using the balance of the residual fuel (approximately 1.74 kilograms) deposited outside the makeup tank and demineralizers as a basis, an additional 10%, or 0.17 kilograms of  $UO_2$ , is assigned to the embedded valves and piping of the MU System.

This results in a fuel estimate of  $2.81 \text{ Kg } UO_2 \pm 27\%$  plus  $0.60 \text{ Kg } UO_2$  (MDL value) for the entire Makeup and Purification System. The range of this estimate extends from 2.05 Kg to 4.17 Kg of  $UO_2$ . See Appendix F for the statistical determination of the range of values. After draindown of the RCS, including the reactor vessel, the MU&P System will be isolated from the RCS by administrative controls to insure that the containment isolation valves are maintained in the closed position.

## 5.0 CONCLUSION

The estimate of record of the amount of  $UO_2$  remaining in the MU&P System is 3.4 kg (at one sigma). Based on the data shown in Table 4 and the data reduction discussion outlined in Appendix F, the range of the estimate of record extends from 2.05 Kg to 4.17 Kg  $UO_2$ . The MDL portion of the upper range value is 0.60 Kg  $UO_2$ .

This estimate of record is derived from existing measurement and sample data, and analyzed values shown for each area listed in Table 4. The MU&P System is expected to remain static since most of the areas have been drained and isolated, and the remaining areas will only be subjected to

low flow rates necessary for final draindown. Additional measurements of the MU&P system are not considered to be warranted based on ALARA considerations and the small quantities of fuel measured to date. After draindown of the RCS, the MU&P System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

## APPENDIX B

### LIQUID RADWASTE DISPOSAL SYSTEM

#### 1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in the Liquid Radwaste Disposal (WDL) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 7, 8, 9, 10, and 11 which show the primary flowpaths between areas containing WDL components and process piping in the AFHB. All statistical uncertainties are expressed as  $\pm$  one (1) sigma limits (defined as one (1) standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the WDL System, and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing WDL system components and process piping. The majority of areas were assayed using gamma spectroscopy featuring either shielded sodium iodide scintillation or high purity germanium detectors connected to multichannel analyzers. Since the presence of Ce-144 (Pr-144) was not always identifiable, MDL calculations were prepared when necessary.

Section 4, "Analysis", explains how the estimate of record of fuel in the WDL System was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the WDL System and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analysis performed.

## 2.0 BACKGROUND

Prior to the accident, the Liquid Radwaste Disposal (WDL) System consisted of process equipment and instrumentation necessary to collect, process, monitor and recycle or dispose of radioactive liquid wastes. All potentially radioactive liquids generated in Unit 2 were collected and processed through the liquid radwaste system prior to recycling or discharge. The liquid radioactive waste was processed on a batch basis to permit optimum control of releases. Aerated drain wastes collected in the reactor drain system, e.g. containment and auxiliary building (AB) drains and laboratory drains, and demineralizer regeneration solutions were segregated based on their origin and activity, and processed through the miscellaneous waste system. The miscellaneous waste system consisted of hold-up tanks, an evaporator, demineralizers, filters, and waste test tanks. Prior to the accident, liquid wastes from normally nonradioactive areas were collected separately and were discharged without treatment into the yard drainage system. The discharge was preceded by routine sampling for radioactivity. If the value exceeded a predetermined level, the liquid waste was processed through the miscellaneous waste system.

The WDL system provided the following functions for the Reactor Coolant (RC) and other liquids:

- a. Boron removal from Reactor Coolant for reactivity control
- b. Collection of all liquid wastes
- c. Hold-up storage of radioactive liquids to allow decay

- d. Process radioactive liquids by ion exchange to remove corrosion and fission products so that its radioactivity level will be as low as practicable
- e. Redundant sampling and monitoring of the liquids before discharging
- f. Controlled discharge of the liquid waste
- g. Recovery of boric acid by evaporation
- h. Volume reduction of the radioactive liquid waste by evaporation
- i. Stripping of radioactive gases
- j. Venting of gases to Waste Gas System
- k. Transfer of concentrated wastes for disposal
- l. Positive control of all liquid releases
- m. Filtering and chemical treatment of liquid releases

During normal plant operation, prior to the 1979 accident, the concentration of boron in the reactor coolant was reduced as the fuel was consumed with core burn-up. Initially the concentration was reduced by simultaneously bleeding highly borated reactor coolant and feeding non-borated coolant into the reactor loops. The highly borated coolant was stored in one of three (3) RC Bleed Holdup Tanks (RCBHT). The non-borated water was fed into the suction of the Makeup (MU) Pumps where it was injected into the reactor loops. Feed and bleed operations were controlled from the Feed and Bleed Panel located in the Control Room.

The RCBHTs collected liquid from the following sources:

- a. Reactor Coolant Letdown Bleed
- b. Reactor Coolant Drain Header
- c. Reactor Coolant Drain Tank
- d. Letdown Line Relief Valve Drain
- e. Reactor Coolant Evaporator Distillate
- f. Makeup Tank Drain

- g. Core Flood Tank Drain
- h. Spent Fuel Cooling System Purification Loop
- i. Evaporator Condensate Test Tanks

Before removal from the RCBHTs, the contents were recirculated to obtain a representative sample for determining disposition of the liquid.

The RC Evaporator was used to concentrate non-volatile radioactive liquid and boric acid. The concentrated liquid was transferred either to the Concentrated Waste Tank (CWT) for disposal or to the Reclaimed Boric Acid Tank for reuse. The volatile and non-condensable gases were removed by a gas stripper and directed to the Radwaste Gas System. The distillate was either pumped through the Evaporator Condensate Demineralizers to the Evaporator Condensate Test Tanks, or it was returned to a RCBHT for storage.

The miscellaneous liquid disposal portion of the WDL System was originally designed to collect, handle, sample and dispose of miscellaneous liquid waste generated throughout the plant. The system tanks and pumps were designed to store the average amount of liquids used or processed in one (1) week. The system was designed to minimize the release of radioactivity to the environment. The discharge paths were monitored for radioactivity and were automatically shut off if radioactivity limits were approached. The miscellaneous liquids disposal portion of the WDL System received liquids from the Service Building, the Auxiliary Building, the Fuel Handling Building and the Reactor Building Sump. These liquids were processed to chemically neutralize and/or reduce the radioactivity level prior to discharge.

The Auxiliary Building (AB) waste system collected liquid from tank drains, reliefs, vents, filter drains, flush line drains, and all the AFHB

floor drains. The liquid wastes were pumped into the AB sump tank. After hold-up, recirculation and sampling, the liquids were normally pumped to the Neutralizer Tanks. After any necessary chemical treatment in the Neutralizer Tank and dependent on the radioactivity level, the liquids were directed to the plant discharge line or storage tanks for long term storage. In summary, all waste liquids were treated so that the solution was neutral or slightly basic.

During the TMI-2 accident, the reactor coolant was letdown from the Reactor Coolant System (RCS) and fuel particles were distributed into many areas of the WDL System. Some of these fuel particles were further distributed into other interconnected systems during the defueling and cleanup operations from 1979 through 1989. A major portion of the fuel remaining in the WDL system is concentrated in large volume containers such as tanks and demineralizers, and is mixed with sediment or demineralizer resins. Other fuel particles are deposited in low velocity areas and dead-ended piping not accessible to flushing action. The primary purposes of the WDL System during the recovery phase was to store water to be used as makeup to the RCS, process stored water through SDS or EPICOR II, and collect water that was letdown from the RCS.

### 3.0 METHODS

Standard gamma spectroscopy methods utilizing NaI (Tl) HPGe detectors were employed to assay the remaining fuel ( $UO_2$ ) in seventeen (17) of the twenty-nine (29) areas occupied by the WDL System. Cerium-144 (Pr-144) was selected as a tracer for the reactor fuel because of its reasonable half-life (284 days), relative high abundance, low escape rate coefficient from reactor fuel and its 2.19 MeV gamma-ray that is readily identifiable on a multichannel analyzer (MCA). The HPGe system was also sometimes able to detect Eu-154 which was used if it was detectable and Ce-144 (Pr-144)

was not. Ce-144 and Eu-154-to-fuel ( $UO_2$ ) ratios were developed using sample data (Reference 6). Details of the specific methods used in each area are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. When a Ce-144(Eu-154) peak was not identifiable, a MDL calculation was performed in accordance with Reference 14. The gross counts in the ROI for Ce-144(Eu-154), converted to an MDL value, was divided by the measured detector efficiency, the calculated photon fluence rate per kilogram of fuel, and the count time.

One (1) area, AX024 utilized a series of gross gamma exposure rate measurements in conjunction with a tungsten-shielded directional probe to quantify the amount of remaining residual fuel. Two (2) other areas, AX129 and AX130 were sampled, and the samples were assayed for fuel content. The residual fuel content in the remaining nine (9) areas was not measured but instead has been determined by analysis of measured fuel deposits in areas with similar flow origin and water processing history. Details of these estimations are discussed in Section 4, Analysis, of this appendix.

#### 4.0 ANALYSIS

The WDL System contains 0.5% of the fuel remaining at TMI-2. The WDL system occupies portions of twenty-nine (29) areas in the Auxiliary and Fuel Handling Buildings (AFHB), which are shown in Figure 7, 8, 9, 10, and 11 except for AX101. The Radwaste Disposal Panel (AX101) is not connected to any of the processing pathways. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan. Twenty (20) of these areas have been assayed for residual fuel, and the results are summarized in Table 5. Seven (7) of the measured areas (AX014, AX015a, AX015b, AX016, AX017, AX018, and AX019) resulted in MDL values. Details of the methods and assumptions used to calculate the



remaining fuel for each area can be found in the engineering calculations listed in Table 5. Since the RC evaporator (AX014) was assayed by Science Applications, Incorporated, their report (Reference 17) is listed as a reference instead of a calculation.

The initial methodology used to estimate the residual fuel ( $UO_2$ ) in the cleanup filters (AX015A,B) and the cleanup demineralizers (AX016 and AX017) estimated a total of 0.317 kg of  $UO_2$  present in all four (4) areas (Reference 18). A review of the data and methods used in the initial calculation indicate that a new approach to data reduction would provide improved and more accurate fuel estimates in the areas of concern. The revised estimates of the residual fuel ( $UO_2$ ) in the pre-filter cleanup filters (WDL-F-6A and 6B) indicated that AX015A contains 0.055 kg of  $UO_2$  and AX015B contains 0.044 kg of  $UO_2$ . Both of these values were MDL values. No measurements of the two (2) cleanup demineralizers (AX016 and AX017) were performed and since fuel particles would not be expected to pass through the cleanup pre-filters; the cleanup demineralizers were estimated to contain no fuel or negligible quantities of fuel ( $UO_2$ ). Similarly, the after-filters (WDL-F-9A and 9B) were estimated to contain no fuel or negligible quantities of fuel. No additional measurements were justified based on the small quantities of fuel estimated and ALARA considerations. A more detailed discussion of the latest methodology is contained in Reference 19.

The remaining seven 7 areas are the Evaporator Condensate Test Tanks and Pumps (AX013), the Radwaste Disposal Control Panel Area (AX101), the RB Sump Pump Filters (AX102), the Instrument and Valve Room (AX128), the Miscellaneous Waste Holdup Tank (AX131), the Miscellaneous Waste Holdup Tank Pump (AX134) and the Neutralizer Tank Filters (FH012). The Evaporator Condensate Test Tanks (AX013) normally collected distillate from the Reactor Coolant Evaporator via one of the Evaporator Condensate

Demineralizers. During the recovery phase for TMI-2, the Evaporate Condensate Test Tanks were used for storage of decontamination water used to clean the walls and floors of contaminated areas in the Reactor, Auxiliary and Fuel Handling Buildings. Prior to storage in the tanks, the water was processed through the EPICOR II System to decontaminate the water by filtration and ion exchange. Recent (1987) water samples and area survey maps show very low radioactivity in AX013. The Monitor Tanks (FH016) have also served to store decontaminated water which was processed through filters and ion exchangers. Recent assay measurements performed in FH106 determined its residual fuel content as 0.003 kg. The Monitor Tanks are the same size, 12,000 gallons each, as the Evaporator Condensate Test Tanks and since both sets of tanks were used in the same capacity during the cleanup phase, it was reasonable to assume they contained equivalent amounts of fuel. Even if the Evaporator Condensate Test Tanks contained double that amount measured in the Monitor Tanks, the resulting difference of 0.003 kg was insignificant in comparison to the uncertainty in the total fuel remaining at TMI-2.

The Radwaste Disposal Control Panel Area (AX101) contained a local Control Panel for the liquid, gaseous and solid radwaste disposal systems. This panel displayed the position of valves, flow rates of mediums being processed, tank levels, tank temperature, tank pressure, and differential pressure indication for all demineralizers and filters. This local control Panel contained only electrically generated signals and consequently was completely isolated from all processing pathways and radioactive materials. Since there was no piping of any kind connected to the local control panel, it was concluded that there was no residual fuel in the Radwaste Disposal Control Panel area.

The Reactor Building (RB) Sump Pump Filter Area (AX102) contained the filters used to filter influent to the Miscellaneous Waste Holdup Tank

(AX131). The RB Sump Pump Filters were used during the TMI-2 accident to filter the water pumped from the flooded RB basement to the Auxiliary Building (AB). Post accident sampling of the sludge in the RB basement found it to contain small quantities of fuel. Therefore, some fuel may have been transferred from the RB basement and deposited in AX102 during the accident. Since the TMI-2 accident, there has been no transfers from the RB basement to the AB sump via the RB sump filters. The RB sump filter elements installed during the accident were removed during 1980 and disposed of as radioactive waste. Subsequent to the accident, the RB sump filters were used routinely to filter water transferred from the AB sump to the Miscellaneous Waste Holdup Tank (MWHT), WDL-T-2. From 1980 to the last change on June 3, 1988, there were over 30 filter change outs of the RB sump filters. Since June 1988, the water processed through the RB sump filters came from the AB sump. Due to the numerous filter change outs, AX102 probably does not contain more fuel than was determined for the AB Sump which was 0.10 kg UO<sub>2</sub>.

The Miscellaneous Waste Holdup Tank (AX131) and the Miscellaneous Waste Holdup Tank (MWHT) Pumps (AX134) are the next components in the processing sequence following the RB sump filters. The next component in the processing sequence is the MWHT pumps which were protected from fuel deposits by the filters and the settling aspects of a large volume tank, MWHT. Considering the processing sequence, it was judged that the total fuel deposits in the RB Sump Pump Filters, the MWHT and the MWHT Pumps could not exceed the fuel measured in the AB Sump (AX012). The AB Sump was a relatively large volume, 7600 gallons, and acts as a settling volume for influent particulates. Based on this settling effect and the numerous filter changes since the accident, it was deduced that there cannot be more fuel deposits downstream from the AB Sump than were actually deposited in the AB Sump itself.

Spectroscopy measurements taken as part of the SNM Accountability Program determined the residual fuel ( $UO_2$ ) remaining in the AB Sump was 0.10 kg ( $\pm$  0.10). Therefore, it was judged for this PDSR that the total fuel remaining in the RB Sump Pump Filter area (AX102), the Miscellaneous Waste Holdup Tank (AX131), and the Miscellaneous Waste Holdup Tank Pump (AX134) was 0.10 kg ( $\pm$  0.10). It is unlikely that transfer of 100% of the fuel could be achieved by simply pumping the AB Sump; the 0.10 kg of fuel appraisal is considered a bounding value.

The Instrument and Valve Room (AX128) contained liquid radwaste processing lines, vent headers for the Radwaste Disposal - Gas System and miscellaneous piping and valves for both systems. The gaseous radwaste disposal system was assumed to contain very little if any fuel deposits because vent gas provided very little driving force to distribute fuel throughout the system. In addition, most venting pathways start out vertically and with very little driving force; transport of significant fuel particulates was highly unlikely. The greater contributor to fuel deposits in AX128 was more likely in the liquid radwaste disposal processing piping which was no longer in use. All of the lines have been flushed and are relatively free of fuel deposits except possibly in the dead-ended piping where flushing was not effective. AX128 is similar to the WDL Valve Room (AX019) but on a much smaller scale. Spectroscopy measurements performed in AX019 resulted in a MDL fuel value of 0.01 kg, and there were many more areas to trap fuel deposits in AX019. Based on the similarity of both areas and that both areas were exposed to similar processing flow conditions during and following the accident, it was concluded that the fuel ( $UO_2$ ) deposit in the Instrument and Valve Room (AX128) was bounded by the 0.01 kg fuel quantity determined for AX019.

The last unmeasured area in the Liquid Radwaste Disposal System was the Neutralizer Tank Filter Area (FH012). The Neutralizer System was used

during the cleanup phase as a batch tank to receive effluent from the MWHT, the AB Sump and the Contaminated Drain Tank, and processed its contents through the EPICOR II System for filtration and purification via ion exchangers. Although originally intended to be used to chemically treat waste liquid, the Neutralizer System had not been used in that manner since the TMI-2 accident. The Neutralizer Tank Filters (WDL-F4A/B) were changed out frequently during their use and were last changed out on June 1, 1990. The spectroscopy measurement performed in the Neutralizer Tank Room (FH009) determined the total residual fuel ( $UO_2$ ) was 0.003 kg. Since the Neutralizer Tank was located upstream from the Neutralizer Tank Filters, and the Neutralizer Tanks acted as a settling volume, and the filters elements were frequently changed out, it was deduced that the fuel deposits in the Neutralizer Tank Filter Area were bounded by the 0.003 kg value for FH009.

Approximately 10% of the processing piping was assumed to be embedded in concrete walls and floors and was never included in the fuel appraisal for the measured areas. Using the balance of the residual fuel (approximately 0.36 kilograms) deposited outside the RC Bleed Holdup Tanks as a basis, an additional 10% or 0.04 kilograms of fuel was assigned to the shielded valves and embedded piping of the WDL system.

This results in a fuel estimate of of 4.13 Kg  $UO_2 \pm 71\%$  plus 0.11 Kg  $UO_2$  (MDL value) for the entire Liquid Radwaste Disposal System. The range of this estimate extends from 1.20 Kg to 7.17 Kg  $UO_2$ . See Appendix F for the statistical determination of the range of values. Since the RCBT measurements were completed, the potential for fuel mobility has been greatly reduced. Water transfer has been limited, and the tank tends to act as a settling area because of the low water velocity inside the tank.

## 5.0 CONCLUSION

The best estimate of record of the amount of  $UO_2$  remaining in the WDL System is 4.2 kg (at one sigma). Based on the data shown in Table 5 and the data reduction discussion outlined in Appendix F, the range of the estimate of record extends from 1.20 kg to 7.17 kg  $UO_2$ . The MDL portion of the upper range is 0.11 kg  $UO_2$ .

This estimate of record is derived from the existing measurement and sample analysis data, and the appraisal values shown for each unmeasured area listed in Table 5. The uncertainty or range is based on the statistical data combination method described in Appendix F. The WDL System is expected to remain static since many of the areas have been drained and isolated, and the remaining areas have been subjected to water transfer on a regular basis. Consequently, all of the fuel that would move has already been relocated to a tank, filter or other low velocity location. Additional measurements of the WDL System are not considered to be justifiable based on ALARA considerations and the small quantities of fuel ( $UO_2$ ) measured to date. After draindown of the RCS, the WDL System will be isolated from the RCS by maintaining the containment isolation valves in a closed position.

## APPENDIX C

### SOLID RADWASTE DISPOSAL SYSTEM

#### 1.0 INTRODUCTION

This appendix presents the analysis of the amount of  $UO_2$  remaining in the Solid Radwaste Disposal (WDS) System contained in the Auxiliary and Fuel Handling Buildings (AFHB). The boundaries of this analysis are illustrated in Figures 12 and 13 which show the primary flowpaths between areas containing WDS components and process piping in the AFHB. All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the WDS System, and the part played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area in the AFHB containing WDS System components and process piping. One area was assayed using gross gamma exposure rate measurements, and a second area was assayed using the standard gamma spectroscopy utilizing a HPGe detector connected to a multichannel analyzer.

Section 4, "Analysis", explains how the estimate of record of fuel in the WDS System was determined and discusses supporting data, assumptions made and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the WDS System and the supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analysis performed.

## 2.0 BACKGROUND

Prior to the accident, the function of the WDS System was to store and transfer bead type resins, concentrated liquid wastes and reclaimed boric acid. The WDS System was composed of two (2) independent subsystems, the resin waste subsystem and the concentrated liquid waste subsystem.

The resin waste subsystem was designed to process bead type ion exchange resins from the demineralizers. The resins exchange ions of radioactive impurities and chemically bind the ions to the resin. Additionally, the resin filters out insoluble radioactive material and activated insoluble corrosion and fission products. The activated corrosion products were formed by corrosion of metallic components and flaking of corroded surfaces which were activated while in transit through the RCS. Fission products formed during fissioning of the uranium fuel entered the reactor coolant by destruction of most of the fuel rod cladding (TMI-2 accident). The corrosion products are insignificant compared with the activity made available through destroyed fuel rod cladding.

Each Spent Resin Storage Tank (WDL-T-1A/B) was designed with a capacity of 516 cubic feet. As designed, the two tanks combined had a nominal two year resin retention capacity. The resin waste subsystem consisted of the Spent Resin Storage Tanks (SRST's), WDS-T-1A and WDS-T-1B, and the Spent Resin Transfer Pump (SRTP), WDS-P-1. The SRTP took suction from either tank and could either recirculate or discharge the contents to storage for off-site disposal.



During the cleanup phase of the Cleanup Program at TMI-2, the resin waste subsystem was used to collect and process spent resins from all the TMI-2 demineralizers and sludge from the RB basement. The greatest quantity of radioactive resins came from the 'A' MU&P demineralizer and was eventually shipped off-site for disposal.

The concentrated liquid waste subsystem was designed to handle liquid wastes which appeared in two basic forms:

- a. Concentrated radioactive boric acid solution
- b. Miscellaneous radioactive liquid wastes.

The activity in both of these liquids resulted from the same sources that challenged the solid system. The concentrated liquid waste subsystem consisted of the Concentrated Waste Tank, WDS-T-2, and the Reclaimed Boric Acid Tank, WDS-T-3. Each tank had its own recirculation/discharge pump and could receive liquids from or have its contents pumped to the Reactor Coolant Evaporator or any other storage tank. The liquid in the Reclaimed Boric Acid Tank was normally pumped to the Boric Acid Mix Tank for reuse. Wastes stored in the Concentrated Waste Tank or the Reclaimed Boric Acid Tank could be reprocessed through the Reactor Coolant Evaporator for further concentration. Normally the wastes were stored to allow radioactivity to decay to a lower level for processing, packaging and subsequent off-site disposal. Due to the intensely radioactive nature of this charge, the WDS System was designed to be controlled remotely. The wastes were sampled to determine their radioactivity and chemical makeup. All operations were batch type and were initiated and terminated by operator action.

During the Cleanup Program at TMI-2, the concentrated liquid waste subsystem was used mostly for storage and solidification of radioactive wastes.

### 3.0 METHODS

Only two (2) areas containing WDS System components and piping were formally measured to determine the remaining fuel ( $UO_2$ ). One (1) area, AX218, which contained the Concentrated Waste Tank, was measured utilizing a HPGe detector. Cerium-144 (Pr-144) was used a tracer for fuel because of its reasonable half-life (284 days), relative high abundance, low escape rate coefficient from fuel and its 2.19 MeV gamma-ray that was readily identifiable on a multichannel analyzer (MCA). A Ce-144-to-fuel ( $UO_2$ ) ratio was previously developed using sample data (Reference 6).

The second area, AX124, which contained the Concentrated Liquid Waste Pump, was measured by using a series of gross gamma exposure rate measurements made with a tungsten-shielded directional probe. Details of the specific methods used for both areas are described in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report. The remaining five (5) unmeasured areas were not formally assayed but were appraised based on their service history and measured fuel deposits in similar areas which were exposed to similar service conditions. Details of these appraisals are discussed in Section 4, Analysis, of this appendix.

### 4.0 ANALYSIS

The WDS System contains less than 0.1% of the fuel ( $UO_2$ ) remaining at TMI-2. The WDS System occupies portions of seven (7) areas in the Auxiliary and Fuel Handling Building (AFHB), which are illustrated in Figures 12 and 13. The area designations are the same as those shown in the TMI-2 Special Nuclear Material (SNM) Accountability Plan (Reference

1). Two (2) of these areas have been measured for residual fuel and the results are summarized in Table 6. The five (5) remaining areas are the two (2) Spent Resin Storage Tanks (WDL-T-1A/B), the Spent Resin Transfer Pump (WDS-P-1), the Reclaimed Boric Acid Tank (WDS-T-3), and the Reclaimed Boric Acid Pump (WDS-P-3). The Spent Resin Storage Tank (SRST) areas, AX008 and AX009, and the Spent Resin Storage Tank Pump area, AX010, were never formally surveyed to determine their fuel content. The piping, tanks, and pump in AX008, AX009 and AX010 were not in the makeup, letdown or waste disposal liquid flowpaths at the time of the accident. However, as a result of resin transfer operations, they probably contain small quantities of fuel. Since the TMI-2 accident, the SRST's were used to collect spent resins from all the TMI-2 demineralizers and were most recently used to collect the resins from the MU&P Demineralizers during 1988. Both SRST's were emptied and flushed early in 1989 and were partially refilled with water for shielding purposes. Their contents were estimated to each contain less than one cubic foot of resin and approximately 600 gallons of water. Since the measured residual fuel values for each MU&P Demineralizer are reliable data, it is assumed for purposes of SNM accountability that any fuel transferred to the SRSTs and dedicated transfer pump has already been accounted for in the MU&P Demineralizer data and no additional quantity is required for the SRSTs and transfer pump. This argument holds true for all the demineralizers at TMI-2. All possible fuel in the SRSTs and transfer pump had to come from a demineralizer (by design) and all demineralizers containing fuel were measured while full of resins; therefore, all possible remaining fuel in the SRSTs and transfer pump has already been included in the residual fuel values established for the various demineralizers. No additional fuel amounts are included in the Summary of Solids Radwaste Disposal System SNM Inventory, Table 6.

The only other unmeasured areas in the WDS System were the Reclaimed Boric

Acid Tank (RBAT), FH010, and the Reclaimed Boric Acid (RBA) Pump, FH011. The RBAT and RBA Pump were never used for their designed function but were used to store accident generated water for a relatively short period. This tank and pump were utilized much less frequently than the Neutralizer Tank, FH009, and Neutralizer Tank Pump, FH008, which were also used to store accident generated water. The Neutralizer Tank and Pump were assayed for  $UO_2$  using gamma spectroscopy and the results were 0.003 kg for the tank and 0.001 kg for the pump. Based on the similarity of service and the short duration of use, it was assumed that the assayed contents for the Neutralizer Tank and Pump were bounding values for the RBAT (FH010) and the RBA Pump (FH011). FH010 and FH011 were assigned residual fuel values of 0.003 kg and 0.001 kg, respectively. Both of these values round off to 0.00 Kg  $UO_2$ .

Approximately 10% of the process piping is assumed to be embedded in concrete walls and floors and was never included in the fuel quantities for the measured areas. Using the total of the determined fuel values for each area in the WDS System as a basis, an additional 10% or 0.00 kilogram of fuel was assigned to the shielded piping of the WDS System.

This results in a fuel estimate of record of 0.01 (rounded to 0.0) kilograms of  $UO_2$  for the entire Solid Waste Disposal System. The range of this estimate extends from 0.00 Kg to 0.02 Kg  $UO_2$ . See Appendix F for the statistical determination of the range of values. It is possible that additional attempts will be made to remove resins from the 'B' Makeup and Purification Demineralizer. In any event, once the accident generated water has been evaporated, the potential for mobility is greatly reduced because only small batches of water will be processed during the post-defueling storage period.

## 5.0 CONCLUSION

The estimate of record of the amount of  $UO_2$  remaining in the WDS System is 0.0 kg (rounded from 0.01 Kg).

This estimate of record is derived from the existing measurements and analyzed data summarized in Table 6. The uncertainty or range is based on the statistical data combination method described in Appendix F. The WDS System is expected to remain static since most of the areas have been drained and isolated, and the remaining areas will not be exposed to flow rates necessary to transport fuel in its present location. It is assumed that all fuel which would normally relocate due to water transport has already been relocated to a tank, filter or other low velocity location. Additional measurements of the WDS System are not considered to be justified based on ALARA considerations and the small quantities of fuel measured to date. After draindown of the RCS, the WDS System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

## APPENDIX D

### BALANCE OF SYSTEMS CONTAINING SNM INVENTORY

#### 1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in selected support systems which with a single exception, contain insignificant quantities of fuel; the Spent Fuel Pool A (FH109) is estimated to contain 3.8 kg of  $UO_2$ . The boundaries of this analysis are illustrated in Figures 14, 15, 16, 17, 18 and 19 which show the primary flowpaths between areas containing system components and the process piping in the AFHB. This appendix addresses the following support systems:

- Defueling Water Cleanup System (DWCS)
- Gas Radwaste Disposal System (WDG)
- Nuclear Sampling System (SNS)
- Spent Fuel Cooling System (SFC)
- Submerged Demineralizer System (SDS)

All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background", describes the original design functions of the individual support systems and the part each system played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods", describes the methods used to assay the residual fuel in each area measured in the AFHB. Only three (3) areas out of the

total of seventeen (17) areas addressed in this appendix were formally measured using a series of gross gamma directional measurements. The balance of the areas were estimated based on their exposure and service history.

Section 4, "Analysis", explains how the estimate of record of fuel in those five (5) systems addressed in this appendix was determined and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion", presents the system total, uncertainty and MDL values for the amount of fuel remaining in the five (5) systems and the supporting rationale leading to the conclusion that the estimate is reasonable based on the available data and analysis performed.

## 2.0 BACKGROUND

Three of the support systems (DWCS, SDS and SNS) were constructed after the accident as part of the TMI-2 Cleanup Program. The other two (2) systems (WDG and SFC) were originally installed when TMI-2 was constructed.

The original primary function of the WDG System was to collect potentially radioactive gas from components and tanks in the plant, compress and deliver this gas to the Waste Gas Decay Tank, store the gas for decay, and recycle or release the gas through the unit vent at a controlled rate within the limits of 10CFR20. The WDG System continued to serve the same function during the cleanup program.

The SFC System was designed to remove the decay heat generated by the spent fuel stored within the fuel storage pools and to purify the water in the transfer canal and fuel storage pools. Due to the short operation of

TMI-2, no spent fuel was stored. The fuel pools were converted and used to store fuel canisters and ion exchange vessels during the cleanup period. Portions of the SFC System were modified during the cleanup program and used for process piping in the DWC System to remove radioactive material from the reactor vessel and fuel storage pools. The DWC System was used primarily to remove organic carbon, radioactive ions, and particulate matter from the fuel transfer canal (FTC), spent fuel pool (SFP) "A", and the reactor vessel (RV). The majority of the particulate matter was removed by processing the water through 0.5 or 16 micron filter canisters. Removal of the radioactive ions (i.e., soluble fission products) and/or organic carbon was accomplished by processing a portion of the filter effluent through 4x4 liners containing zeolite or charcoal materials that were similar to those used for EPICOR II.

The DWC System was composed of two (2) major subsystems, the RV cleanup system and the FTC/SFP cleanup system. Only a small portion of the RV cleanup system was located in Fuel Handling Building and that portion, the ion exchangers, was used on a very infrequent basis. The entire FTC/SFP cleanup system was located in the Fuel Handling Building, elevation 347'. The primary function of the two (2) subsystems was to remove suspended solids down to the nominal 0.5 micron rating to maintain the clarity of the water. Sample analysis data of RCS particulates indicated that approximately 1% of the sample was 0.5 microns or less (Reference 20). Since a filter-aid material was used to improve the filtering efficiency of the main filters, it was estimated that substantially less than 1% of the particulates passed through the main filters for possible deposit downstream from the filters. Most of the effluent from the main filters was returned to its source, RV or FTC/SFP "A". Up to 15 GPM was directed to the ion exchangers to remove fission products and/or organic carbon. Two (2) additional filters were installed on each side of the ion exchangers to remove any particulates greater than 0.45 microns that might possibly



have entered the ion exchanger loop. All of the filter canisters have been removed from the TMI site.

The SDS System was a temporary liquid radwaste processing system installed in the TMI-2 Spent Fuel Pool "B" and the area immediately adjacent to the spent fuel pool. The system was designed to remove the fission products contained in the Reactor Containment Building and sump, the RCS and other liquid systems such as WDL, etc. by the process of ion exchange. The SDS System consisted of a liquid waste processing system, an off gas system, a monitoring and sampling system, and a solid waste handling system. The liquid waste processing system decontaminated the RCS water by filtration and demineralization. The off gas system collected, filtered and adsorbed radioactive gases produced during processing, sample dewatering, and spent SDS liner venting. The solid waste handling system also provided for moving, dewatering, storing and loading of filter and demineralizer vessels into the shipping cask. SDS performed the notable task of decontaminating the Reactor Building Sump water (625,000 gallons of water generated during the accident), the RCS (90,000 gallons), the FTC and water accumulated in the RCBT's or MWHT in combination with EPICOR II. The SDS System has been permanently shutdown although final flushing/drainage has not yet been accomplished.

The SNS System was installed after the 1979 TMI-2 accident to provide a means to obtain representative liquid and gas samples from selected points containing post-accident waste. The sampling system provided a location for remote sampling of the sources of radioactive water within TMI-2. This eliminated the necessity of running highly radioactive sample water to the TMI-1 Sample Room and made the sampling operation independent of TMI-1 operations. The SNS System provided a means for recirculation and purging of the sample lines, backflushing with demineralized water to reduce activity levels, and transferring the sampling waste back to the

storage system. The temporary sample hood for TMI-2 was located in the north end of the FHB at elevation 305'. This location permitted connection to the existing sampling lines running through the FHB. Samples from the individual subsystems could be collected by in-line sample containers or taken as grab samples. In addition, a boronometer was provided to monitor the boron concentration in the RCS. A separate loop, with its own pump and tank, was provided for calibration of the boronometer. The sample hood and room were ventilated by a filtered ventilation system which withdrew air from the hood/room and exhausted it to the Auxiliary Building ventilation system. A sample sink was provided to collect spillage from all operations conducted in the hood and also collected the liquid used to flush the sample lines from the hood isolation valves to the sample collection point. All sample waste was transferred to either the MWHT (WDL-T-2) or the RCHTs 'B' or 'C' (WDL-T-1B or 1C).

### 3.0 METHODS

Only three (3) of the seventeen (17) areas containing SNS, SDS, WDG, DWCS and SFC System components and piping were formally measured to assay the residual fuel. The three (3) areas, FH105, FH106 and FH109, were measured using a series of gross gamma exposure rate measurements in conjunction with a tungsten-shielded directional probe. Details of the specific procedures used for all three (3) areas is contained in Section 3, Measurement Methods, of the basic AFHB Post-Defueling Survey Report and in each of the engineering calculations listed in Table 7. The remaining fourteen (14) unmeasured areas were analyzed based on their service history and their similarity to other areas that were measured. Details of these appraisals are discussed in Section 4, Analysis, of this appendix.

#### 4.0 ANALYSIS

The results of the three (3) measured areas, FH105, FH106 and FH109, are shown in Table 7. Over 99% of the remaining fuel in the seventeen (17) areas addressed in this appendix is contained in FH109, Spent Fuel Pool A (SFPA). Most of this fuel was transported to SFPA from the RV as debris adherent to the outside of the fuel bearing canisters. The SFPA was used to store and to flush the outsides of fuel canisters prior to loading into the shipping casks. The balance of the Spent Fuel Cooling (SFC) System areas (AX118, AX119, AX120 and FH111) contain an insignificant quantity of fuel because the system was never exposed to the RCS primary cooling water and was not operated during the cleanup program at TMI-2. The only SFC System areas that were utilized during the cleanup program were the Spent Fuel Pool A (FH109), Spent Fuel Pool B (FH110), and a portion of the fill/drain piping for the FTC and pools.

SFPA was measured using gamma spectroscopy, and the results indicate that 3.8 kg of fuel ( $UO_2$ ) was still deposited in SFPA. The other pool, SFPB, was used to house SDS vessels and remained clean through the entire recovery program. The water in the SFPB has been processed through EPICOR II and/or SDS and subsequently was not recontaminated. SDS vessels installed in SFPB were stored in secondary containment enclosures to collect any leakage from the connections to the vessels. Pool water was continuously drawn through these enclosures and passed through separate ion exchangers. This design prevented the pool water from becoming contaminated. The leakage was monitored through the established SDS sampling system. A residual fuel value of 0.003 kg (rounded to 0.00) was arbitrarily assigned to SFPB (FH110) to cover any potential film deposits that might remain in the connecting hoses which are stored in the FH110. FH111 is another relatively small storage pool which communicated with

FH110 but did not house equipment processing radioactive water or possible fuel. Since FH111 is clean and was not used to store containers which contained core debris, it was concluded that there was no residual fuel in FH111. FH302 is an area directly above the SFPB where SDS vessels were located and contained the supporting equipment for operating the SDS System. Numerous hoses and instrumentation are located in FH302, and the hoses probably have developed thin film deposits of fuel. A residual value of 0.001 kg UO<sub>2</sub> (rounded to 0.00) was assigned to FH302. The other areas in the Spent Fuel Cooling System were the Spent Fuel Cooler Area (AX118), the Spent Fuel Demineralizer Area (AX119), and the Spent Fuel Filter Area (AX120). They were never operated during or after the accident and therefore, were not internally exposed to core debris. AX118 served a second function during the cleanup program which was to store and package radioactive waste material. AX118 was periodically contaminated and decontaminated to prevent high level buildup.

The Gaseous Radwaste Disposal (WDG) System occupies a major portion of the following areas: AX113, AX125, AX126 and AX127. These four (4) areas contained the Waste Gas Filter (WDG-F-1), two (2) Waste Gas Decay Tanks (WDG-T-1A/1B), and the Waste Gas Analyzer. The waste gas was filtered prior to entering the Waste Gas Analyzer and it was assumed that no fuel is in the analyzer itself. On the other hand, there is some chance that a small amount of fuel was blown into AX125, AX126 or AX127. On this basis, a residual fuel value of 0.001 kg UO<sub>2</sub> (rounded to 0.00) was assigned to each of the three areas.

The only areas remaining (FH205, FH303 and FH305) were utilized for portions of the Defueling Water Cleanup System (DWCS). As a consequence, it was assumed that a thin film of fuel residue was retained in the hoses and pipes used to process the RCS and FTC/SFPA water. An arbitrary value of 0.002 kg UO<sub>2</sub> was assigned to FH205 and 0.001 kg UO<sub>2</sub> was assigned to

FH303 and FH305. These values were in line with the sample line values measured in FH106.

This results in a fuel estimate of record of 3.8 kilograms of  $UO_2$  for all of the systems listed in Table 7. The range of this estimate extends from 0.30 Kg to 5.09 Kg  $UO_2$ . See Appendix F for the statistical determination of the range of values. After draindown of the RCS, including the reactor vessel, the Gaseous Radwaste Disposal System, the Spent Fuel Cooling System, the temporary Nuclear Sampling System, the Defueling Water Cleanup System, and the Submerged Demineralizer System will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

## 5.0 CONCLUSION

The estimate of record of the amount of  $UO_2$  remaining in the Balance of Systems Containing SNM Inventory, listed in Table 7, is 3.8 kg. See Appendix F for details of statistical data combination.

This estimate of record was derived from the measurement and analysis results summarized in Table 7. These areas are expected to remain static since most of the areas have been drained and isolated, and the remaining areas have been subjected to water flows on a regular basis. Consequently, all of the fuel that would move has already been relocated to a tank, filter or other low-velocity location. Additional measurements of those areas listed in Table 7 are not considered to be justifiable based on ALARA considerations and the small quantities of fuel ( $UO_2$ ) measured to date. After draindown of the RCS, the areas will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

## APPENDIX E

### AFHB AREAS THAT CONTAIN NO SNM

#### 1.0 INTRODUCTION

This appendix presents the analysis of the amount of fuel remaining in a miscellaneous group of areas which could contain quantities of fuel if their normal function had been misused. The following is a list of those miscellaneous areas.

<u>Area</u>	<u>Description</u>	<u>Primary System</u>
AX001	RB Emergency Booster Pumps	RB Emergency Cooling (RR)
AX109	NS Coolers and Pumps	Nuclear Service Closed Cooling Water (NS)
AX110	Intermediate Coolers	Interm. Closed Cooling Water (IC)
AX111	Interm. Cooling Pumps & Filters	Interm. Closed Cooling Water (IC)
AX212	DH Surge Tank	DH Closed Cooling Water (DC)
AX214	Decon Facility	Temporary Cleaning Facility
AX219	Atmospheric Monitors	Health Physics (HP)
AX220	Caustic Liquid Mixing Tanks	Chemical Addition (CA)
AX221	Caustic Liquid Mixing Tanks	Chemical Addition (CA)
AX402	Cooling Water Surge Tanks	IC, RB and NS Systems
FH005	MDH Vault Pumps	Misc. Decay Heat Removal (MDH)
FH006	DH Service Coolers	Decay Heat Closed Cooling Water (DH)
FH007	Access Area - North	Fuel Handling Building
FH103	Sample Room	Temporary Nuclear Sampling (SNS)
FH107	Trash Compactor	Temporary Storage Area

FH108	Truck Bay	Fuel Canister Loading
FH203	Off-gas Separator & Standpipe	Submerged Demin. System (SDS)
FH204	Tanks & Charging Pumps	Standby Pressure Control System
FH301	Fuel Handling Bridge	Fuel Handling System

All statistical uncertainties are expressed as  $\pm$  one sigma limits (defined as one standard deviation) unless otherwise stated.

Section 2, "Background," describes the original design functions of all of the primary systems and the part each system played during the 1979 TMI-2 accident and the TMI-2 Cleanup Program.

Section 3, "Methods," is not applicable to this appendix because no areas were formally assayed; they were analyzed for fuel based on their vulnerability during the accident and service during the TMI-2 Cleanup Program.

Section 4, "Analysis," explains how the estimate of record of fuel for those areas addressed in this appendix was determined and discusses assumptions made and conclusions drawn.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of fuel remaining in these areas and the supporting rationale leading to the conclusion that the estimate is reasonable based on the analysis performed.

## 2.0 BACKGROUND

Six (6) of the support systems, SNS, SDS, SPC, MDH, temporary cleaning facility, and a modified fuel handling bridge, were constructed after the accident and installed as part of the TMI-2 Cleanup Program. The other

six (6) systems, CA, DC, HP, IC, NS and RR, were originally installed when TMI-2 was constructed. A short discussion of the original function of each system follows.

The Chemical Addition (CA) System was designed to dissolve chemicals in demineralized water. Some of the chemicals mixed were boric acid, sulfuric acid, sodium thiosulfate, lithium hydroxide, and hydrazine. The only chemical used during the TMI-2 Cleanup Program was the boric acid to maintain the RCS boron level to protect against criticality.

The original function of the Decay Heat Closed Cooling Water (DC) System was to supply demineralized cooling water to the DH Removal Coolers and Pumps which supplied emergency cooling to the RCS and the core. The DC System remained operable to provide cooling of the DH Removal System pumps, motors and coolers in the event it ever became necessary to operate the DH Removal System.

The original function of the Intermediate Closed Cooling Water (IC) System was to provide cooling water to various equipment located inside the Reactor Building. Some of the equipment supplied cooling water were the letdown coolers, the reactor coolant pumps, the control rod drive mechanism, and the SG hot drain coolers. Since none of this equipment operated after the accident, there was no need to operate the IC System during the cleanup program. The system has remained shutdown and isolated for the last eight years.

The original function of the Nuclear Service Closed Cooling Water (NS) System was to provide cooling water to the following equipment: instrument air compressors, RC pumps and motors, waste gas compressors, spent fuel pools, RC evaporator, RB spray pumps, MU pumps, NS pumps and motors, and the RB emergency cooling booster pumps. Since most of this



equipment was operating or available to operate in any emergency, the NS System continued to operate during all of the cleanup program.

The original function of the RB Emergency Cooling (RR) System was to provide cooling water for all nuclear-related and fuel handling requirements. The RR System also provided emergency cooling to the RB cooling coils via the RB Emergency Cooling Booster Pumps. No emergency cooling has been required since the TMI-2 accident.

The original objective of the Health Physics (HP) Program was to minimize the radiation exposure of station personnel and visitors. During the TMI-2 Cleanup Program, it was even more important because TMI-2 was severely contaminated during the 1979 accident. Initially, the airborne radioactivity monitoring system (atmospheric monitors) was comprised of nine (9) fixed channels located in AX219 and two (2) mobile channels, each consisting of a particulate, iodine and gas measured monitor in series. During the cleanup program, additional mobile monitors were utilized as necessary to prevent worker overexposure.

The Mini Decay Heat Removal (MDH) System was installed after the 1979 accident to provide a flow path for make-up water from the Borated Water Storage Tank (BWST) to the RCS. The MDH System was never operated but was held in abeyance for emergency situations requiring additional cooling of the RCS.

The Submerged Demineralizer System (SDS) was a temporary liquid radwaste processing system installed in the 328' and 347' elevations of the Fuel Handling Building. The system was designed to remove the fission products contained in the RB sump and other liquid radwaste systems by the process of ion exchange. SDS was used to decontaminate the RB sump water (625,000 gallons), the RCS (90,000 gallons), and the FTC in combination with EPICOR

II. The SDS has been disassembled and all of the ion exchange and filter vessels have been removed from the FHB.

The Temporary Nuclear Sampling (SNS) System was installed after the 1979 TMI-2 accident to collect liquid and gas samples from selected points. The SNS System provided a location for remote sampling of the sources of radioactive water within TMI-2. This eliminated the necessity of running highly radioactive sample water to the TMI-1 Sample Room and made the sampling function independent of TMI-1.

A temporary cleaning facility was constructed at the north end of the Auxiliary Building (AX214) to decon tools and radiation monitors. The facility was used to clean equipment that was slightly contaminated and remained relatively clean throughout its use.

The Fuel Handling Bridge had been removed from the FHB shortly after the accident. The Fuel Handling Bridge was modified to transport fuel canisters prior to reinstallation. A completely new canister handling trolley was added to the existing Fuel Handling Bridge. The new trolley included the canister transfer shield, grapple, grapple guiding tool, hoists, cable and hose reels, and a load cell with a digital readout. Shielding for the fuel canisters was provided by a fixed, shielded mast (i.e., canister transfer shield) attached to the trolley. A nine (9) foot high shield collar was located on the lower end of the canister transfer shield.

The Standby Pressure Control (SPC) System was a recovery system used primarily as a source of borated water for makeup to the RCS. The SPC System was normally isolated from the RCS except when it was injecting borated water into the RCS. During defueling operations with the reactor vessel head removed, the SPC System was used as a source of makeup water

to the RCS or a supply of flush water to the DWCS Systems. The SPC System components were located in the new fuel storage cell (FH204) on the 331' elevation of the FHB.

### 3.0 METHODS

All nineteen (19) areas addressed in this appendix were analyzed based on their vulnerability during the TMI-2 accident and their service history during the TMI-2 Cleanup Program. Since none of the areas were assumed to contain fuel, no formal assay measurements were performed. Details of the individual analyses are discussed in Section 4, Analysis, of this appendix.

### 4.0 ANALYSIS

Since no areas were formally measured and it was assumed that no fuel was located in any of the areas, this section is a discussion of the history of each area and the reasons why it has been concluded that they do not contain fuel.

AX001 (RB Emergency Cooling Booster Pumps), AX109 (Nuclear Service Coolers and Pumps), AX110 (Intermediate Coolers), AX111 (Intermediate Cooling Pumps and Filters), AX212 (DH Surge Tank), AX402 (IC and NS Cooling Water Surge Tanks), and FH006 (DH Service Coolers) all contain components which are part of secondary cooling systems.

All of these components are separated from radioactivity by heat exchangers or cooling tubes which give positive protection against migrating fuel. Area survey data support the supposition there is no fuel in any of these areas, even though some of these areas contain various degrees of contamination.

AX220 and AX221 are areas set aside for mixing chemicals prior to injection into various process systems. The areas were designed to facilitate mixing of boric acid, sulfuric acid, sodium thiosulfate, lithium hydroxide, and hydrazine. During the cleanup program, only boric acid was mixed to keep the RCS water borated to approximately 5,000 ppm while defueling the reactor vessel. Both areas were slightly contaminated but were never exposed to fuel. All injection lines to fuel bearing systems were protected by double valve isolation and were checked every 24 hours.

AX219 contains the Atmospheric Radiation Monitors for the Station Vent, RB Purge and Vent, and the Hydrogen Purge. All of these monitors received contamination from time to time and had to be decontaminated; they were never directly exposed to fuel.

FH005 contained two MDH pumps which were never placed in service; operation was limited to testing of the system. These pumps were designed to inject borated water into the RCS but were never used. Their flow pathway remained isolated during the entire cleanup program and was never exposed to fuel.

FH203 contained the off-gas standpipe and separator of the SDS which were used intermittently to process vent gas from the ion exchanger vessels. The gas flow was filtered prior to being exhausted to the plant vent stack. These filters prevented any particulates from entering the off-gas standpipe and separator and prevented fuel from reaching FH203.

FH103 contains portions of the Nuclear Sampling System. It is possible that some samples contained fuel. The entire sample area was cleaned and decontaminated periodically to remove any fuel that may have collected in the area. Upon completion of analyses, all samples were transferred to

either the MWHT or the RCBHT 'B' or 'C'. Therefore, any fuel that might have been located in FH103 would have eventually found its way to either the MWHT or the RCBHT 'B' or 'C'.

AX214 was a decon facility installed during the cleanup program to decon hand tools and radiation monitors. The facility was used to clean slightly contaminated equipment and remained relatively clean throughout its use. All contaminated cleaning materials were disposed through proper channels, and there was no chance of fuel remaining in the facility. Area surveys were performed periodically and showed very low radiation exposure rates, less than 0.2 mR/hr, which supports the conclusion that no fuel was deposited in AX214.

FH204 contained the SPC charging pumps and tanks which were used to supply borated water to the RCS and other flushing operations. The system was used intermittently to supply water to the DWCS Systems. When not supplying water at high pressure, the SPC System was isolated from the RCS by double valve isolation which was verified every 24 hours. Since this was a new system maintained at a positive pressure when not isolated, there was no chance that fuel could have entered FH204.

FH301 contained the fuel handling bridge which was used to move fuel canisters in the SFPA. Since the canisters were contaminated with fuel when they entered the SFPA, it was possible that fuel was deposited in the canister transfer shield. The transfer shield was stored in the SFPA, and any residual fuel remaining was measured as part of the total amount determined for the fuel pool (FH109).

All of these above areas have been evaluated for fuel content, and it has been concluded that it is improbable any remaining fuel not already analyzed and accounted for elsewhere, exists in any of those areas.

## 5.0 CONCLUSION

The estimate of record of the amount of  $UO_2$  remaining in the nineteen (19) areas listed in Section 1.0, Introduction, is 0.0 kg. This estimate of record is based on the discussion outlined in Section 4.0, Analysis. Additional measurements of any of the areas discussed in this appendix are not considered to be justifiable based on ALARA considerations and the absence of measured fuel ( $UO_2$ ) to date. After draindown of the RCS, the areas will be isolated from the RCS by maintaining the containment isolation valves in their closed position.

## APPENDIX F

### CONSTRUCTION OF OVERALL ERROR

The total  $UO_2$  deposited within the AFHB is expressed as a sum. The total fuel value, termed the estimate for record, is the sum of all quantities measured or deduced within the Auxiliary and Fuel Handling Building areas. However problems arise in the expression of the overall uncertainty for differently derived individual uncertainties.

Three cases exist for the present. Most of the uncertainty is expressed as the usual bi-directional one sigma uncertainty. Due to unmeasurably small quantities, several results were expressed as minimum detectable quantities at the 95% confidence interval. Finally one area has an unsymmetrical uncertainty expressed as one (1) sigma. The purpose of this appendix is to detail the method used to combine the three cases into an overall uncertainty.

The method employed here was borrowed from suggestions made by Dr. Tingey, Idaho State University as follows:

Consider a set of counting data  $x_i$ ,  $i = 1, 2, \dots, k$ , which are intended to estimate fuel quantities  $Z_i$ ,  $i = 1, 2, \dots, k$ . Assume the variance of  $Z_i$ ,  $\sigma_i^2$  is known.

Consider another set  $y_j$ ,  $j = 1, 2, \dots, \ell$  for which the fuel quantity say  $\theta_j$  can only be less than some minimum detectable level  $m_j$  and greater than or equal to zero.

Consider

$$W = \sum_{i=1}^k x_i + \sum_{j=1}^{\ell} y_j \quad (1)$$

The expected fuel value of  $W$ , say  $N$ , is

$$N = \sum_{i=1}^k Z_i + \sum_{j=1}^{\ell} \theta_j \quad (2)$$

$$N + \left[ \sum_{i=1}^k \sigma_i^2 \right]^{1/2} \geq R \geq N - \left[ \sum_{i=1}^k \sigma_i^2 \right]^{1/2} - \sum_{j=1}^l \theta_j \quad (3)$$

Note that the limits are not symmetrical about

$N$

Also note that when combining these limits with others determined similarly, a partitioning over all measurements being combined should be made in accordance with the above and equation (3) applied to the total set, that is we cannot simply combine the upper limits or lower limits statistically.

where:

- $x_i$  are the measured counting data
- $Z_i$  are the expected fuel values corresponding to the measured counting data
- $\sigma_i^2$  is the variance of  $Z_i$
- $y_j$  are the MDL counting data
- $\theta_j$  are the fuel values which are known to lie between zero and  $\theta_j$
- $W$  is the sum of  $x_i$ 's and  $y_j$ 's
- $N$  is the expected fuel value of  $W$
- $R$  is the range of the expected fuel values

#### UO<sub>2</sub> in Areas with Symmetrical Uncertainty

As described in the body of the report, fuel is deposited in three systems. They are the Makeup and Purification, MU&P, the Liquid Radwaste Disposal, WDL, and the Solid Radwaste Disposal, WDS, Systems. Results for the individual areas comprising the MU&P and the WDL systems are presented in Tables F1, and F2. Results for



the Solid Waste Disposal System were taken from the body of the report since only normal symmetrical uncertainty was found.

Summary results are presented in two parts. Total symmetrical uncertainty is presented in Table F3, and the combined overall uncertainty is discussed later.

The first step is to define the total fuel quantity for values measured above background with symmetrical uncertainties. The combined error was defined, in quadrature, as the square root of the sum of squares of the individual uncertainties for each area. The MU&P and WDL system totals are also shown on Tables F1 and F2.

### Total UO<sub>2</sub> in All Areas

Table F3 provides the total fuel quantity and uncertainty for all increments with symmetrical uncertainty. Two additional components must be included to derive the grand total and overall uncertainty. The first is the simple sum of all fuel values derived from quantities too small to be measured above background, MDL's. The last is the fuel value for FH109 of 3.80 kg with an associated asymmetric uncertainty of +34% and -92%.

The grand total, also the fuel quantity of record, was calculated by equation 2 as:

$$\text{Total} = 6.95 + 0.71 + 3.80 = 11.46 \text{ kg UO}_2$$

The overall range of values was calculated by a form of equation 3 modified to include the asymmetric FH109 uncertainties, the negative term on the left and the positive on the right as follows:

$$\text{low value} = (6.95 - 3.03) + 3.80*(1-.92) = 4.22 \text{ kg UO}_2$$

and:

$$\text{high value} = (6.95 + 3.03) + 3.80*(1+.34) + 0.71 = 15.78 \text{ kg UO}_2$$

The estimate for record is 11.5 kg UO<sub>2</sub> with an uncertainty of +38% and - 63%.

TABLE F-1

FUEL (UO<sub>2</sub>) AND UNCERTAINTY CALCULATIONS FOR THE MU&P SYSTEM

<u>Area</u>	<u>Fuel Quantity</u>	<u>Uncertainty</u>	<u><math>\sigma</math></u>	<u><math>\sigma^2</math></u>
AX004	0.03 Kg	± 58%	0.0174	0.000303
AX005	0.00 Kg	-	-	-
AX114	1.06 Kg	± 44%	0.4664	0.217529
AX115	0.13 Kg	± 25%	0.0325	0.001056
AX116	0.31 Kg	± 95%	0.2945	0.086730
AX117	0.06 Kg	± 68%	0.0408	0.001665
FH001	0.46 Kg	± 85%	0.3910	0.152881
FH003a	0.01 Kg	± 100%	0.0100	0.000100
FH003b	0.10 Kg	± 100%	0.1000	0.010000
FH002 )				
FH004 )	0.16 Kg	± 50%	0.0800	0.006400
FH014 )				
FH101	0.32 Kg	± 71%	0.2272	0.051620
Misc.	<u>0.17 Kg</u>	± 95%	0.1615	<u>0.026082</u>
Totals	2.81 Kg			0.555104
				Square Root = 0.7450
				Percentage = 26.51 or 27%

The sum of the MDL values is 0.60 Kg.

The MU&P System total is 2.81 Kg ± 27% + 0.60 Kg.

TABLE F-2

FUEL (UO<sub>2</sub>) AND UNCERTAINTY CALCULATIONS FOR THE WDL SYSTEM

<u>Area</u>	<u>Fuel Quantity</u>	<u>Uncertainty</u>	<u><math>\sigma</math></u>	<u><math>\sigma^2</math></u>
AX012	0.10 Kg	± 104%	0.1040	0.010816
AX020	3.50 Kg	± 83%	2.9050	8.439025
AX021	0.31 Kg	± 46%	0.1426	0.020335
AX024	0.02 Kg	± 36%	0.0072	0.000052
AX102 } AX131 } AX134 }	0.10 Kg	± 104%	0.1040	0.010816
AX128	0.01 Kg	± 95%	0.0095	0.000090
AX501	0.01 Kg	± 76%	0.0076	0.000058
AX502	0.01 Kg	± 59%	0.0059	0.000035
AX503	0.01 Kg	± 100%	0.0100	0.000100
AX504	0.01 Kg	± 92%	0.0092	0.000085
FH112	0.01 Kg	± 55%	0.0055	0.000030
Misc	0.04 Kg	± 95%	0.0380	0.001444
Totals	4.13 Kg			8.482886

Square Root = 2.9125  
 Percentage = 70.52 or 71%

The sum of the MDL values is 0.11 Kg.

The WDL System total is 4.13 Kg ± 71% + 0.11 Kg.

TABLE F-3

FUEL SUMMARY FOR AUXILIARY AND FUEL HANDLING BUILDING

Symmetrical Data:

<u>System</u>	<u>Fuel Quantity</u>	<u>Uncertainty</u>	<u><math>\sigma</math></u>	<u><math>\sigma^2</math></u>
MU&P	2.81 Kg	± 27%	0.7587	0.575626
WDL	4.13 Kg	± 71%	2.9323	8.598383
WDS	0.01 Kg	± 57%	0.0057	0.000032
Totals	6.95 Kg			9.174041
			Square Root =	3.0289
			Percentage =	43.58 or 44%

Asymmetrical Data:

Two (2) other components must be included to determine the total quantity of fuel remaining in the AFHB. Those are the asymmetric term and the MDL term. The asymmetric value for FH109 is 3.80 Kg UO<sub>2</sub> +34%, -92%. The MDL value is 0.71 Kg UO<sub>2</sub>.